U.S. NUCLEAR REGULATORY COMMISSION

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GUIDANCE FOR DEVELOPING PRINCIPAL DESIGN CRITERIA FOR NON-LIGHT-WATER REACTORS

A. INTRODUCTION

Purpose

This regulatory guide (RG) describes the Nuclear Regulatory Commission's (NRC's) proposed guidance on how the general design criteria (GDC) in Appendix A, "General Design Criteria for Nuclear Power Plants," of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," (Ref. 1), may be adapted for non-light-water reactor (non-LWR) designs. This guidance may be used by non-LWR reactor designers, applicants, and licensees to develop principal design criteria (PDC) for any non-LWR designs, as required by the applicable NRC regulations, for nuclear power plants. The RG also describes the NRC's proposed guidance for modifying and supplementing the GDC to develop PDC that address two specific non-LWR design concepts: sodium-cooled fast reactors (SFRs), and modular high temperature gas-cooled reactors (MHTGRs).

Applicability

This RG applies to nuclear power reactor designers, applicants, and licensees of non-LWR designs subject to 10 CFR Part 50 and 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (Ref. 2)[1].

Applicable Regulations

- 10 CFR Part 50 provides regulations for licensing production and utilization facilities.
    - o 10 CFR Part 50, Appendix A, contains the GDC that establish the minimum requirements for the PDC for water-cooled nuclear power plants. Appendix A also establishes that the GDC are generally applicable to other types of nuclear power units and are intended to provide guidance in determining the PDC for such other units.

[1] While the design criteria described in this RG were developed for nuclear power reactor applicants developing non-LWR designs, the design criteria described in this RG may be applied, as appropriate, to non-light-water non-power reactors.

Written suggestions regarding this guide or development of new guides may be submitted through the NRC's public Web site in the NRC Library at http://www.nrc.gov/reading-rm/doc-collections/, under Document Collections, in Regulatory Guides, at http://www.nrc.gov/reading-rm/doc-collections/reg-guides/contactus.html,

Electronic copies of this RG, previous versions of this guide, and other recently issued guides are also available through the NRC's public Web site in the NRC Library at http://www.nrc.gov/reading-rm/doc-collections/, under Document Collections, in Regulatory Guides. This RG is also available through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html, under ADAMS Accession Number (No.) ML17325A611. The regulatory analysis may be found in ADAMS under Accession No. ML16330A179. The associated draft guide DG-1330 may be found in ADAMS under Accession No. ML16301A307, and the staff responses to the public comments on DG-1330 may be found under ADAMS Accession No. ML17325A616.

- 10 CFR 50.34(a)(3)(i) requires that an application for a construction permit include the PDC for a proposed facility.

- 10 CFR Part 52 governs the issuance of early site permits, standard design certifications, combined licenses, standard design approvals, and manufacturing licenses for nuclear power facilities.

  - 10 CFR 52.47(a)(3)(i) requires that an application for a design certification include the PDC for a proposed facility.

  - 10 CFR 52.79(a)(4)(i) requires that an application for a combined license include the PDC for a proposed facility.

  - 10 CFR 52.137(a)(3)(i) requires that an application for a standard design approval include the PDC for a proposed facility.

  - 10 CFR 52.157(a) requires that an application for a manufacturing license include the PDC for a proposed facility.

Related Guidance, Communications, and Policy Statements

- NUREG-1338, "Draft Preapplication Safety Evaluation Report for the Modular High-Temperature Gas-Cooled Reactor (MHTGR)," issued December 1995, provides the NRC staff's review and insights on the MHTGR design (Ref. 3).

- NUREG-1368, "Preapplication Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Liquid Metal Reactor," issued February 1994, provides the NRC staff's review and insights on the design for the GE-Hitachi PRISM liquid-metal reactor (LMR) (Ref. 4).

- NUREG-0968, "Safety Evaluation Report Related to the Construction of the Clinch River Breeder Reactor Plant," issued March 1983, provides the staff's evaluation of the Clinch River construction permit application (Ref. 5).

- NUREG-1369, "Preapplication Safety Evaluation Report for the Sodium Advanced Fast Reactor (SAFR) Liquid-Metal Reactor," issued December 1991, provides the NRC staff's review and insights on the SAFR design (Ref. 6).

- SECY-93-092, "Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and their Relationship to Current Regulatory Requirements," dated April 8, 1993, provides staff insights on issues pertaining to advanced designs and proposes resolutions (Ref. 7).

- SRM-SECY-93-092, "Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and their Relationship to Current Regulatory Requirements," issued July 30, 1993, provides the Commission position on topics discussed in SECY-93-092 (Ref. 8).

- SECY-03-0047, "Policy Issues Related to Licensing Non-Light-Water Reactor Designs," dated March 28, 2003, provides, for Commission consideration, options and recommended positions for resolving the seven policy issues associated with the design and licensing of future non-LWR designs (Ref. 9).

- SRM-SECY-03-0047, "Policy Issues Related to Licensing Non-Light-Water Reactor Designs," issued June 26, 2003, provides the Commission position on the topics discussed in SECY-03-0047 (Ref. 10).

- NRC, "Next Generation Nuclear Plant—Assessment of Key Licensing Issues," dated July 17, 2014, provides the NRC staff's review and insights on the Next Generation Nuclear Plant MHTGR proposed licensing approach (Ref. 11).

- NRC, "Policy Statement on the Regulation of Advanced Reactors" (73 FR 60612, October 14, 2008), establishes the Commission's expectations related to advanced reactor designs to protect the environment and public health and safety and promote the common defense and security with respect to advanced reactors (Ref. 12).

Purpose of Regulatory Guides

The NRC issues RGs to describe to the public methods that the staff considers acceptable for use in implementing specific parts of the agency's regulations, to explain techniques that the staff uses in evaluating specific problems or postulated events, and to provide guidance to applicants. Regulatory guides are not substitutes for regulations and compliance with them is not required. Methods and solutions that differ from those set forth in RGs will be deemed acceptable if they provide a basis for the findings required for the issuance or continuance of a permit or license by the Commission.

Paperwork Reduction Act

This RG provides guidance for implementing the mandatory information collections in 10 CFR Parts 50 and 52 that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et. seq.). These information collections were approved by the Office of Management and Budget (OMB), under control numbers 3150-0011 and 3150-0151. Send comments regarding this information collection to the Information Services Branch, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by e-mail to Infocollects.Resource@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0011, 3150-0151) Office of Management and Budget, Washington, DC 20503.

Public Protection Notification

The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number.

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B. DISCUSSION

Reason for Issuance

This revision (Revision 0) provides guidance for developing PDC for non-LWRs. Applications for a construction permit, design certification, combined license, standard design approval, or manufacturing license are required by 10 CFR 50.34(a)(3)(i), 10 CFR 52.47(a)(3)(i), 10 CFR 52.79(a)(4)(i), 10 CFR 52.137(a)(3)(i), and 10 CFR 52.157(a), respectively, to include the PDC for the facility in their applications.

Background

The NRC Regulatory Framework

In accordance with its mission, the NRC protects public health and safety 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 guidance. The applicable regulatory requirements are found in Chapter I of Title 10, "Energy," of the *Code of Federal Regulations*, 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 RGs, interim staff guidance, standard review plans, NUREGs, review standards, and Commission policy statements. These regulatory requirements and guidance represent the entirety of the regulatory framework that an applicant should consider when preparing an application for review by the NRC. A key part of the regulatory requirements is in the general design criteria (GDC) in Appendix A to 10 CFR Part 50. These high-level GDC requirements support the design of the current nuclear power plants and are addressed in 10 CFR 50.34, "Contents of Applications; Technical Information." Because the current GDC are based on LWR technology, the NRC developed the non-LWR design criteria, included as appendices to this RG, to provide guidance for developing PDC for non-LWR technology.

The nuclear power plants presently operating in the United States were licensed under the process described in 10 CFR Part 50. The NRC and its predecessor, the U. S. Atomic Energy Commission (AEC), approved construction permits for these plants between 1964 and 1978 and granted the most recent operating license under 10 CFR Part 50 in 2015. The regulations in 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. The NRC applied some of these new regulations retroactively to operating reactors while applying others only to new reactors.

The NRC used its experience in licensing nuclear power plants to develop 10 CFR Part 52, which it issued in 1989 and has used for the most recent new reactor licensing reviews, reactor design certifications, and early site permits. The regulations in 10 CFR Part 52 apply lessons learned from licensing the operating reactors, provide an alternative to the current process described in 10 CFR Part 50, and increase the 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 CFR Part 52, rather than those in 10 CFR Part 50. For this reason, 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 (Ref. 13), 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" (Ref. 14), 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. In the staff requirements memorandum (SRM) to SECY-15-0002 (Ref. 15), the Commission approved the staff's recommendation to revise the regulations in 10 CFR Part 50 and Part 52 for new power reactor applications to reflect lessons learned from recent new reactor licensing activities and to more closely align with each other. This RG is not intended to be an accompaniment to the aforementioned rulemaking.

Role of the General Design Criteria in the Regulatory Framework

As mentioned above, the GDC contained in Appendix A to 10 CFR Part 50 are an important part of the NRC's regulatory framework. For LWRs, they provide minimum requirements for PDC, which establish the necessary 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 are also intended to provide guidance in establishing the PDC for non-LWRs. The GDC serve as the fundamental criteria for the NRC staff when reviewing the SSCs that make up a nuclear power plant design particularly when assessing the performance of their intended safety functions in design basis events postulated to occur during normal operations, anticipated operational occurrences (AOOs), and postulated accidents. All production and utilization facilities licensed under 10 CFR Part 50, including both LWRs and non-LWRs, are required to describe PDC in their preliminary safety analysis report supporting a construction permit application as described in 10 CFR 50.34(a)(3).

NRC Policy on Advanced Reactors

From the NRC staff's regulatory perspective, the characteristics of an "advanced reactor" have 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 on advanced nuclear power reactors, "Policy Statement on the Regulation of Advanced Reactors," which included items to be considered in their designs. The Commission's 2008 policy statement reinforced and updated the policy statements on 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 nuclear power 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, the NRC regulations and policies may be addressed in a different manner than for traditional LWRs.

Role of the General Design Criteria for Non-LWRs

As discussed in Section A of this RG, applications for a construction permit, design certification, combined license, standard design approval, or manufacturing license, respectively, must include the PDC for the facility. The PDC for light water nuclear power reactors are derived from the GDC in Appendix A to 10 CFR Part 50.

Title 10 CFR 50.34[2] states:

> "Appendix A to 10 CFR part 50, general design criteria (GDC), establishes minimum requirements for the principal design criteria for watercooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the Commission and provides guidance to applicants in establishing principal design criteria for other types of nuclear power units."

Appendix A to 10 CFR part 50 states:

> "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."

Together, these requirements recognize that different requirements may need to be adapted for non-LWR designs and that the GDC in 10 CFR 50 Appendix A are not regulatory requirements for non-LWR designs but provide guidance in establishing the PDC for non-LWR designs. The non-LWR design criteria developed by the NRC staff and included in Appendices A to C of this regulatory guide are intended to provide stakeholders with insight into the staff's views on how the GDC could be interpreted to address non-LWR 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, 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 non-LWR design criteria are an important first step to address the unique characteristics of non-LWR technology. The NRC recognizes the future benefits to risk informing the non-LWR design criteria to the extent possible, depending on the design information and data available. The NRC's "Vision and Strategy: Safely Achieving Effective and Efficient Non-Light-Water Reactor Mission Readiness" (Ref. 16), outlines mid- and long-term activities to develop, as necessary, a risk-informed, performance-based non-LWR regulatory framework. Implementing the mid- and long-term Implementation Action Plans as part of the Vision and Strategy activities will help NRC determine whether risk informed non-LWR design criteria should be included as part of a new regulatory framework.

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2 Similar language is included in 10 CFR 52.47(a)(3)(i), 10 CFR 52.79(a)(4)(i), 10 CFR 52.137(a)(3)(i), and 10 CFR 52.157(a).

DOE-NRC Initiative Phase 1

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 non-LWR technologies—specifically, to address the existing GDC, which may 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 the PDC that address non-LWR design features while recognizing that the underlying safety objective of each GDC still applies.

The assessment of the GDC with respect to non-LWR designs was accomplished in two phases. Phase 1 was managed by a team including representatives of the DOE 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-LWR technologies (i.e., sodium-cooled fast reactors (SFRs), lead fast reactors (LFRs), gas-cooled fast reactors (GCRs), modular high-temperature gas-cooled reactors (MHTGRs), fluoride high-temperature reactors (FHRs), and molten-salt reactors (MSRs)). Using this information, DOE then reviewed the existing NRC GDC to determine their applicability to 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." DOE submitted this report to the NRC for consideration in December 2014 (Ref. 17). In it, DOE proposed a set of advanced reactor design criteria (ARDC), which could serve the same purpose for non-LWRs as the GDC serve for LWRs. The ARDC are intended to be technology inclusive to align with the six technologies above. In addition to the technology-inclusive ARDC, DOE proposed two sets of technology-specific, non-LWR design criteria. These criteria are intended to apply to SFRs and MHTGRs and are referred to as the SFR design criteria (SFR-DC) and the MHTGR design criteria (MHTGR-DC), respectively. DOE developed the technology-specific design criteria to demonstrate how the GDC could be adapted to specific technologies in which there was some level of maturity and documented design information available.[3] DOE determined that the safety objectives for some of the current GDC did not address design features specific to SFR and MHTGR technologies (e.g., sodium or helium coolant, passive heat removal systems, etc.). Additional design criteria were developed to address unique features of those designs.

DOE-NRC Initiative Phase 2

After DOE issued its report in December 2014, an NRC multidisciplinary team was assembled to review the report, other pertinent references, and NRC documents, such as NUREGs, reports, and white papers. The NRC held a public meeting on January 21, 2015, to discuss the report with DOE and to describe NRC's plans to develop regulatory guidance for non-LWR reactor design criteria (Ref. 18).

During its review, the NRC staff formulated questions and clarifications necessary to obtain a full understanding of the 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 following documents contain the NRC questions and DOE responses:

- "NRC Staff Questions on the DOE Report, 'Guidance for Developing Principal Design Criteria for Advanced Non-Light Water Reactors,'" dated June 5, 2015, and "Response to NRC Staff

[3] The technology-specific design criteria were developed using available design information, previous NRC pre-application reviews of the design types, and more recent industry and DOE national laboratory initiatives in these technology areas (see Reference 17). It is the responsibility of the designer or applicant to provide and justify the PDC for a specific design.

Questions on the U.S. Department of Energy Report, 'Guidance for Developing Principal Design Criteria for Advanced Non-Light Water Reactors,'" dated July 15, 2015 (Ref. 19 for both), and

- "Questions on the U.S. Department of Energy Report, 'Guidance for Developing Principal Design Criteria for Advanced Non-Light Water Reactors,'" dated August 17, 2015, and "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,'" dated September 15, 2015 (Ref. 20 for both).

After consideration of the DOE report, DOE responses to NRC staff questions, and other applicable information relevant to the NRC regulatory philosophy and current understanding of non-LWR designs, the NRC developed its own version of the ARDC, SFR-DC, and MHTGR-DC. While reviewing the DOE report, NRC staff considered whether to develop one generic set of non-LWR design criteria or to follow the DOE model and develop the technology specific design criteria as well. After considering the diversity of the design features for the two mature technologies, the NRC staff chose to develop the SFR-DC and MHTGR-DC in addition to the ARDC.

The NRC issued a draft version of design criteria for informal public comment titled, "Public Comment Sought - Advanced Non-Light Water Reactor Design Criteria," on April 7, 2016 (Ref. 21). The NRC staff noted in the introductory material of this invitation that comments received would not be responded to individually but would be considered by the NRC staff when developing the draft RG. By June 8, 2016 the NRC received over 350 public comments from over 20 stakeholder organizations (Ref. 22). NRC used the informal public comments and discussions during the public meeting held on October 11, 2016 (Ref. 23), to develop DG-1330, "Guidance for Developing Principal Design Criteria for Non-Light Water Reactors," NRC staff issued the draft RG on February, 2017 (Ref. 24), for a 60 day comment period. NRC staff received over 120 comments on DG-1330 (Ref. 25), and held a public meeting on August 24, 2017 to discuss topics that warranted additional public interaction (Ref. 26). The tables in Appendices A, B, and C of this RG represent the staff's final version of the design criteria that incorporates many of the public comments.

Key Assumptions and Clarifications Regarding the non-LWR Design Criteria

The NRC staff applied the following key assumptions when developing the non-LWR design criteria:

- The underlying safety objectives of the GDC still apply.

- ARDC, SFR-DC, and MHTGR-DC apply to normal operations, anticipated operational occurrences, and postulated accidents (design basis).

- The NRC has regulations and orders on severe accidents and beyond-design-basis events (BDBEs) for LWRs. Similar regulations for non-LWRs were not defined as part of this initiative. The current regulations may or may not be applicable to non-LWRs. It is the responsibility of the applicant to demonstrate compliance with applicable severe accident and BDBE regulations and orders, demonstrate why any that are not applicable do not apply, and demonstrate how other design specific severe accidents or BDBE that can occur will be mitigated.

- While developing the non-LWR design criteria, the staff assumed that a core disruptive accident will be demonstrated to be a severe accident or a BDBE by the applicant. A core disruptive accident would result in a loss of a coolable geometry such that multiple non-LWR design criteria would be violated.

- Safety design approach for non-LWRs can differ substantially from those associated with LWRs.

- Proposed GDC adaptations were focused on those needed for improved regulatory certainty and clarity.

- The NRC intends the ARDC to apply to the six advanced reactor technology types identified in the DOE report; however, in some instances, one or more of the criteria from the SFR-DC or MHTGR-DC may be more applicable to a design or technology than the ARDC.

- MHTGR refers to the category of HTGRs that use the inherent high temperature characteristics of tristructural isotropic (TRISO) coated fuel particles, graphite moderator, and helium coolant, as well as passive heat removal from a low power density core with a relatively large height-to-diameter ratio within an uninsulated steel reactor vessel. The MHTGR is designed in such a way to ensure that during design basis events (including loss of forced cooling or loss of helium pressure conditions) radionuclides are retained at their source in the fuel and regulatory requirements for offsite dose are met at the exclusion area boundary.

- The SFR-DC and MHTGR-DC were developed because the designs were mature and the design features diverse for these technologies. Additional sets of technology-specific design criteria (e.g., MSRs, LFRs) may be developed in the future as more information about the designs becomes available.

- Some of the concepts discussed in the RG are policy issues that may require NRC Commission review and approval. Examples are functional containment performance requirements and the use of specified acceptable system radionuclide release design limits in place of specified acceptable fuel design limits. The NRC has not had the opportunity to fully consider these as they are specific to non-LWR designs.

- Non-LWR designs should provide enhanced margins of safety when compared to LWRs. They may use simplified, passive, or other innovative design features to accomplish their safety and security functions.

Harmonization with International Standards

The International Atomic Energy Agency (IAEA), in collaboration with the International Project on Innovative Nuclear Reactors and Fuel Cycles and the Generation IV International Forum, established the Sodium-Cooled Fast Reactor Task Force. The SFR Task Force is collaborating with international designers, government organizations, and regulators to develop safety design criteria and safety design guidelines for SFRs. The IAEA also has a Coordinated Research Activity on MHTGR safety design criteria.

The NRC will continue to monitor and collaborate on these documents and consider using them to the extent practical in developing SFR design criteria. The NRC will follow its standard procedures for public participation in the development of future NRC documents that reference or endorse international standards.

C. STAFF REGULATORY GUIDANCE

This section contains information on the intended use of the RG. It also contains NRC staff's determination of the applicability of each GDC to the non-LWR design criteria. This is illustrated in the table titled, "Table 1: Non-Light-Water Reactor Crosswalk." The actual ARDC, SFR-DC, and MHTGR-DC and NRC staff technology-specific rationale for adaptions to the GDCs to develop the PDC are contained in Appendices A—C to this RG.

Intended Use of This Regulatory Guide

This RG provides guidance to reactor designers, applicants, and licensees of non-LWR designs for developing PDC[4]. Since the GDC in 10 CFR 50 Appendix A are not regulatory requirements for non-LWR designs but provide guidance in establishing the PDC for non-LWR designs, non-LWR applicants would not need to request an exemption from the GDC in 10 CFR Part 50 when proposing PDC for a specific design.

Applicants may use this RG to develop all or part of the PDC and are free to choose among the ARDC, SFR-DC, or MHTGR-DC to develop each PDC after considering the underlying safety basis for the criterion and evaluating the rationale for the adaptation described in this RG. For example, FHRs are molten salt reactors that use TRISO fuel, which is the same fuel used for MHTGR technologies. An FHR designer could use the MHTGR-DC where appropriate for the design. Another example is the MSRs that use liquid fuel. An MSR designer may need to develop new PDC for liquid fuel and systems to support this design.

In each case, it is the responsibility of the designer or applicant to provide not only the PDC for the design but also supporting information that justifies to the NRC how the design meets the PDC submitted, and how the PDC demonstrate adequate assurance of safety. In instances where a GDC or non-LWR design criterion (ARDC, SFR-DFC, and MHTGR-DC) is not proposed, the designer/applicant must provide a basis and justify the omission from a safety perspective.

As noted earlier in this RG under the subheading, "Role of the General Design Criteria for Non-LWRs," the current GDC are regulations and therefore use the words "shall" and "must" that are appropriate for regulatory requirements. The proposed ARDC, SFR-DC, and MHTGR-DC presented in Appendices A, B, and C to this RG also use the words "shall" and "must" for consistency with the GDC, and so that non-LWR applicants can use them in the same manner as GDC when developing PDC. However, this wording is not intended to imply that they are regulatory requirements, as they are contained in a guidance document.

Finally, the non-LWR design criteria as developed by the NRC staff are intended to provide stakeholders with insights into the staff's views on how the GDC could be interpreted to address non-LWR design features; however, these are not considered to be final or binding on what may eventually be required from a non-LWR applicant.

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4 While the design criteria described in this RG were developed for nuclear power reactor applicants developing non-LWR designs, the design criteria described in this RG may be applied, as appropriate, to non-light-water non-power reactors.

Non-LWR Crosswalk Table

The following table (Table 1) provides a summary and crosswalk between the LWR GDC contained in 10 CFR Part 50 Appendix A and the NRC staff's determination of their applicability to the ARDC, SFR-DC, and MHTGR-DC. For each design criterion, the table denotes the status (same as GDC, same as ARDC, modified for ARDC, modified for SFR-DC, or modified for MHTGR-DC). Table 1 also uses redline-strikeout to identify the design criteria titles that have been modified for non-LWRs. Words removed from the title are in **red** with a strikethrough and words that have been added are in **blue** and underlined. The actual ARDC, SFR-DC, and MHTGR-DC and NRC staff technology-specific rationale for adaptions to the GDCs are contained in Appendices A—C to this RG.

The table consists of five columns:

Column 1—Criterion Number
Column 2—Current GDC Title (from 10 CFR Part 50, Appendix A)
Column 3—ARDC Title/Status (showing conformity to or deviation from 10 CFR Part 50, Appendix A)
Column 4—SFR-DC Title/Status (showing conformity to or deviation from 10 CFR Part 50, Appendix A)
Column 5—MHTGR-DC Title/Status (showing conformity to or deviation from 10 CFR Part 50, Appendix A)

The table is divided into seven sections similar to those in 10 CFR Part 50, Appendix A:

Section I—Overall Requirements (Criteria 1–5)
Section II—Multiple Barriers (Criteria 10–19)
Section III—Reactivity Control (Criteria 20–29)
Section IV—Fluid Systems (Criteria 30–46) for ARDCs, and SFR-DC
Section IV —Heat Transport Systems (Criteria 30-46) for MHTGR-DC
Section V—Reactor Containment (Criteria 50–57)
Section VI—Fuel and Radioactivity Control (Criteria 60–64)
Section VII—Additional SFR-DC (Criteria 70–77) and Additional MHTGR-DC (Criteria 70–72)

TABLE 1: NON-LIGHT-WATER-REACTOR CROSSWALK

I. Overall Requirements

Criterion	Current GDC Title	ARDC Title/Status	SFR-DC Title/Status	MHTGR-DC Title/Status
1	*Quality standards and records.*	Same as GDC	Same as GDC	Same as GDC
2	*Design bases for protection against natural phenomena.*	Same as GDC	Same as GDC	Same as GDC
3	*Fire protection.*	*Fire protection.* Modified for ARDC	Same as ARDC	Same as ARDC
4	*Environmental and dynamic effects design bases.*	*Environmental and dynamic effects design bases.* Modified for ARDC	*Environmental and dynamic effects design bases.* Modified for SFR-DC	*Environmental and dynamic effects design bases.* Modified for MHTGR-DC
5	*Sharing of structures, systems, and components.*	Same as GDC	Same as GDC	Same as GDC

II. Multiple Barriers

Criterion	Current GDC Title	ARDC Title/Status	SFR-DC Title/Status	MHTGR-DC Title/Status
10	*Reactor design.*	Same as GDC	Same as GDC	*Reactor design.* Modified for MHTGR-DC
11	*Reactor inherent protection.*	*Reactor inherent protection.* Modified for ARDC	Same as ARDC	Same as ARDC
12	*Suppression of reactor power oscillations.*	*Suppression of reactor power oscillations.* Modified for ARDC	Same as ARDC	*Suppression of reactor power oscillations.* Modified for MHTGR-DC
13	*Instrumentation and control.*	*Instrumentation and control.* Modified for ARDC	*Instrumentation and control.* Modified for SFR-DC	*Instrumentation and control.* Modified for MHTGR-DC

TABLE 1: NON-LIGHT-WATER-REACTOR CROSSWALK

II. Multiple Barriers				
Criterion	**Current GDC Title**	**ARDC Title/Status**	**SFR-DC Title/Status**	**MHTGR-DC Title/Status**
14	*Reactor coolant pressure boundary.*	*Reactor coolant* ~~**pressure**~~ *boundary.* Modified for ARDC	***Primary*** *coolant* ~~**pressure**~~ *boundary.* Modified for SFR-DC	*Reactor* ***helium*** ~~**coolant**~~ *pressure boundary.* Modified for MHTGR-DC
15	*Reactor coolant system design.*	*Reactor coolant system design.* Modified for ARDC	***Primary*** ~~**Reactor**~~ *coolant system design.* Modified for SFR-DC	*Reactor* ***helium pressure boundary*** ~~**coolant**~~ *design.* Modified for MHTGR-DC
16	*Containment design.*	Same as GDC	*Containment design.* Modified for SFR-DC	*Containment design.* Modified for MHTGR-DC
17	*Electric power systems.*	*Electric power systems.* Modified for ARDC	Same as ARDC	*Electric power systems.* Modified for MHTGR-DC
18	*Inspection and testing of electric power systems.*	*Inspection and testing of electric power systems.* Modified for ARDC	Same as ARDC	Same as ARDC
19	*Control room.*	*Control room.* Modified for ARDC	*Control room.* Modified for SFR-DC	Same as ARDC

III. Reactivity Control				
Criterion	**Current GDC Title**	**ARDC Title/Status**	**SFR-DC Title/Status**	**MHTGR-DC Title/Status**
20	*Protection system functions.*	Same as GDC	Same as GDC	*Protection system functions.* Modified for MHTGR-DC
21	*Protection system reliability and testability.*	Same as GDC	Same as GDC	Same as GDC
22	*Protection system independence.*	Same as GDC	Same as GDC	Same as GDC

TABLE 1: NON-LIGHT-WATER-REACTOR CROSSWALK

III. Reactivity Control				
Criterion	**Current GDC Title**	**ARDC Title/Status**	**SFR-DC Title/Status**	**MHTGR-DC Title/Status**
23	*Protection system failure modes.*	Same as GDC	*Protection system failure modes.* Modified for SFR-DC	Same as GDC
24	*Separation of protection and control systems.*	Same as GDC	Same as GDC	Same as GDC
25	*Protection system requirements for reactivity control malfunctions.*	*Protection system requirements for reactivity control malfunctions.* Modified for ARDC	Same as ARDC	*Protection system requirements for reactivity control malfunctions.* Modified for MHTGR-DC
26	*Reactivity control system redundancy and capability.*	*Reactivity control systems* ~~*redundancy and capacity*~~ Modified for ARDC	Same as ARDC	*Reactivity control systems* Modified for MHTGR-DC
27	*Combined reactivity control systems capability*	~~**Combined reactivity control systems capability**~~ DELETED and incorporated into ARDC 26	Same as ARDC	Same as ARDC
28	*Reactivity limits.*	*Reactivity limits.* Modified for ARDC	*Reactivity limits.* Modified for SFR-DC	*Reactivity limits.* Modified for MHTGR-DC
29	*Protection against anticipated operational occurrences.*	Same as GDC	Same as GDC	Same as GDC

TABLE 1: NON-LIGHT-WATER-REACTOR CROSSWALK

IV. Fluid Systems (Heat Transport Systems for MHTGRs)				
Criterion	**Current GDC Title**	**ARDC Title/Status**	**SFR-DC Title/Status**	**MHTGR-DC Title/Status**
30	*Quality of reactor coolant pressure boundary.*	*Quality of reactor coolant ~~pressure~~ boundary.* Modified for ARDC	*Quality of ~~reactor~~ **primary** coolant ~~pressure~~ boundary.* Modified for SFR-DC	*Quality of reactor **helium** ~~coolant~~ pressure boundary.* Modified for MHTGR-DC
31	*Fracture prevention of reactor coolant pressure boundary.*	*Fracture prevention of reactor coolant ~~pressure~~ boundary.* Modified for ARDC	*Fracture prevention of ~~reactor~~ **primary** coolant ~~pressure~~ boundary.* Modified for SFR-DC	*Fracture prevention of reactor **helium** ~~coolant~~ pressure boundary.* Modified for MHTGR-DC
32	*Inspection of reactor coolant pressure boundary.*	*Inspection of reactor coolant ~~pressure~~ boundary.* Modified for ARDC	*Inspection of ~~reactor~~ **primary** coolant ~~pressure~~ boundary.* Modified for SFR-DC	*Inspection of reactor **helium** ~~coolant~~ pressure boundary.* Modified for MHTGR-DC
33	*Reactor coolant makeup.*	*Reactor coolant **inventory maintenance** ~~makeup~~.* Modified for ARDC	*~~Reactor~~ **Primary** coolant **inventory maintenance** ~~makeup~~.* Modified for SFR-DC	Not applicable to MHTGR.
34	*Residual heat removal.*	*Residual heat removal.* Modified for ARDC	*Residual heat removal.* Modified for SFR-DC	*Residual heat removal.* Modified for MHTGR-DC
35	*Emergency core cooling.*	*Emergency core cooling.* Modified for ARDC	Same as ARDC	Not applicable to MHTGR.
36	*Inspection of emergency core cooling system.*	*Inspection of emergency core cooling-system.* Modified for ARDC	Same as ARDC	*Inspection of **passive** ~~emergency core cooling~~ **residual heat removal** system.* Modified for MHTGR-DC
37	*Testing of emergency core cooling system.*	*Testing of emergency core cooling-system.* Modified for ARDC	Same as ARDC	*Testing of **passive residual heat removal** ~~emergency core cooling~~ system.* Modified for MHTGR-DC
38	*Containment heat removal.*	*Containment heat removal.* Modified for ARDC	Same as ARDC	Not applicable to MHTGR.

TABLE 1: NON-LIGHT-WATER-REACTOR CROSSWALK

IV. Fluid Systems (Heat Transport Systems for MHTGRs)				
Criterion	**Current GDC Title**	**ARDC Title/Status**	**SFR-DC Title/Status**	**MHTGR-DC Title/Status**
39	*Inspection of containment heat removal system.*	*Inspection of containment heat removal system.* Modified for ARDC	Same as ARDC	Not applicable to MHTGR.
40	*Testing of containment heat removal system.*	*Testing of containment heat removal system.* Modified for ARDC	Same as ARDC	Not applicable to MHTGR.
41	*Containment atmosphere cleanup.*	*Containment atmosphere cleanup.* Modified for ARDC	Same as ARDC	Not applicable to MHTGR.
42	*Inspection of containment atmosphere cleanup systems.*	Same as GDC	Same as GDC	Not applicable to MHTGR.
43	*Testing of containment atmosphere cleanup systems.*	*Testing of containment atmosphere cleanup systems.* Modified for ARDC	Same as ARDC	Not applicable to MHTGR.
44	*Cooling water.*	***Structural and equipment cooling.*** ~~**Cooling water**~~ Modified for ARDC	Same as ARDC	***Structural and equipment cooling.*** ~~***Cooling water***~~ Modified for MHTGR-DC
45	*Inspection of cooling water system.*	*Inspection of* ***structural and equipment*** cooling ~~***water***~~ systems. Modified for ARDC	Same as ARDC	Same as ARDC
46	*Testing of cooling water system.*	*Testing of* ***structural and equipment*** cooling ~~***water***~~ *systems.* Modified for ARDC	Same as ARDC	Same as ARDC

TABLE 1: NON-LIGHT-WATER-REACTOR CROSSWALK

V. Reactor Containment				
Criterion	**Current GDC Title**	**ARDC Title/Status**	**SFR-DC Title/Status**	**MHTGR-DC Title/Status**
50	*Containment design basis.*	*Containment design basis.* Modified for ARDC	*Containment design basis.* Modified for SFR-DC	Not applicable to MHTGR.
51	*Fracture prevention of containment pressure boundary.*	*Fracture prevention of containment pressure boundary.* Modified for ARDC	*Fracture prevention of containment pressure boundary.* Modified for SFR-DC	Not applicable to MHTGR.
52	*Capability for containment leakage rate testing.*	*Capability for containment leakage rate testing.* Modified for ARDC	*Capability for containment leakage rate testing.* Modified for SFR-DC	Not applicable to MHTGR.
53	*Provisions for containment testing and inspection.*	*Provisions for containment testing and inspection.* Modified for ARDC	*Provisions for containment testing and inspection.* Modified for SFR-DC	Not applicable to MHTGR.
54	*Piping systems penetrating containment.*	*Piping systems penetrating containment.* Modified for ARDC	*Piping systems penetrating containment.* Modified for SFR-DC	Not applicable to MHTGR.
55	*Reactor coolant pressure boundary penetrating containment.*	*Reactor coolant ~~pressure~~ boundary penetrating containment.* Modified for ARDC	*~~Reactor~~ Primary coolant ~~pressure~~ boundary penetrating containment.* Modified for SFR-DC	Not applicable to MHTGR.
56	*Primary containment isolation.*	*~~Primary~~ Containment isolation.* Modified for ARDC	*~~Primary~~ Containment isolation.* Modified for SFR-DC	Not applicable to MHTGR.
57	*Closed system isolation valves.*	*Closed system isolation valves.* Modified for ARDC	*Closed system isolation valves.* Modified for SFR-DC	Not applicable to MHTGR.

TABLE 1: NON-LIGHT-WATER-REACTOR CROSSWALK

VI. Fuel and Radioactivity Control				
Criterion	**Current GDC Title**	**ARDC Title/Status**	**SFR-DC Title/Status**	**MHTGR-DC Title/Status**
60	*Control of releases of radioactive materials to the environment.*	Same as GDC	Same as GDC	Same as GDC
61	*Fuel storage and handling and radioactivity control.*	*Fuel storage and handling and radioactivity control.* Modified for ARDC	Same as ARDC	Same as ARDC
62	*Prevention of criticality in fuel storage and handling.*	Same as GDC	Same as GDC	Same as GDC
63	*Monitoring fuel and waste storage.*	Same as GDC	Same as GDC	Same as GDC
64	*Monitoring radioactivity releases.*	*Monitoring radioactivity releases.* Modified for ARDC	*Monitoring radioactivity releases.* Modified for SFR-DC	*Monitoring radioactivity releases.* Modified for MHTGR-DC

TABLE 1: NON-LIGHT-WATER-REACTOR CROSSWALK

VII. Additional Technology-Specific Design Criteria				
Criterion	**Current GDC Title**	**ARDC Title/Status**	**SFR-DC Title/Status**	**MHTGR-DC Title/Status**
70	*N/A*	*N/A*	*Intermediate coolant system.*	*Reactor vessel and reactor system structural design basis.*
71	*N/A*	*N/A*	*Primary coolant and cover gas purity control.*	*Reactor building design basis.*
72	*N/A*	*N/A*	*Sodium heating systems.*	*Provisions for periodic reactor building inspection.*
73	*N/A*	*N/A*	*Sodium leakage detection and reaction prevention and mitigation.*	*N/A*
74	*N/A*	*N/A*	*Sodium/water reaction prevention/mitigation.*	*N/A*
75	*N/A*	*N/A*	*Quality of the intermediate coolant boundary.*	*N/A*
76	*N/A*	*N/A*	*Fracture prevention of the intermediate coolant boundary.*	*N/A*
77	*N/A*	*N/A*	*Inspection of the intermediate coolant boundary.*	*N/A*
78	*N/A*	*N/A*	*Primary coolant system interfaces.*	*N/A*
79	*N/A*	*N/A*	*Cover gas inventory maintenance.*	*N/A*

D. IMPLEMENTATION

The purpose of this section is to provide information on how applicants and licensees[5] may use this guide and information regarding the NRC's plans for using this RG. In addition, it describes how the NRC staff complies with 10 CFR 50.109, "Backfitting," and any applicable finality provisions in 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants."

Use by Applicants and Licensees

Applicants and licensees may voluntarily[6] use the guidance in this document to demonstrate compliance with the underlying NRC regulations. Methods or solutions that differ from those described in this RG may be deemed acceptable if the applicant or licensee provides sufficient basis and information for the NRC staff to verify that the proposed alternative demonstrates compliance with the appropriate NRC regulations.

Licensees may use the information in this RG for actions which do not require NRC review and approval such as changes to a facility design under 10 CFR 50.59, "Changes, Tests, and Experiments." Licensees may use the information in this RG or applicable parts to resolve regulatory or inspection issues.

Use by NRC Staff

The NRC staff does not intend or approve any imposition or backfitting of the guidance in this RG. The NRC staff does not expect any existing licensee to use or commit to using the guidance in this RG, unless the licensee makes a change to its licensing basis. The NRC staff does not expect or plan to request licensees to voluntarily adopt this RG to resolve a generic regulatory issue. The NRC staff does not expect or plan to initiate NRC regulatory action which would require the use of this RG without further backfit consideration. Examples of such unplanned NRC regulatory actions include: issuance of an order requiring the use of the RG, requests for information under 10 CFR 50.54(f) as to whether a licensee intends to commit to use of this RG, or generic communication, or promulgation of a rule requiring the use of this RG.

During regulatory discussions on plant-specific operational issues, the staff may discuss with licensees various actions consistent with staff positions in this RG, as one acceptable means of meeting the underlying NRC regulatory requirement. Such discussions would not ordinarily be considered backfitting. And, unless this RG is part of the licensing basis for a facility, the staff may not represent to the licensee that the licensee's failure to comply with the positions in this RG constitutes a violation.

If an existing licensee voluntarily seeks a license amendment or change and (1) the NRC staff's consideration of the request involves a regulatory issue directly relevant to this new RG and (2) the specific subject matter of this RG is an essential consideration in the staff's determination of the acceptability of the licensee's request, then the staff may request that the licensee either follow the guidance in this RG or provide an equivalent alternative process that demonstrates compliance with the

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5 In this section, "licensees" refers to licensees of nuclear power plants under 10 CFR Parts 50 and 52; the term "applicants," refers to applicants for licenses and permits for (or relating to) nuclear power plants under 10 CFR Parts 50 and 52, and applicants for standard design approvals and standard design certifications under 10 CFR Part 52.

6 In this section, "voluntary" and "voluntarily" mean that the licensee is seeking the action of its own accord, without the force of a legally binding requirement or an NRC representation of further licensing or enforcement action.

underlying NRC regulatory requirements. This is not considered backfitting as defined in 10 CFR 50.109(a)(1) or a violation of any of the issue finality provisions in 10 CFR Part 52.

Additionally, an existing applicant may be required to comply with new rules, orders, or guidance if 10 CFR 50.109(a)(3) applies.

If a licensee believes that the NRC is either using this RG or requesting or requiring the licensee to implement the methods or processes in this RG in a manner inconsistent with the discussion in this Implementation section, then the licensee may file a backfit appeal with the NRC in accordance with the guidance in NUREG-1409, "Backfitting Guidelines" (Ref. 27), and the NRC Management Directive 8.4, "Management of Facility-Specific Backfitting and Information Collection" (Ref. 28).

ACRONYMS/ABBREVIATIONS

AEC	Atomic Energy Commission
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	anticipated operational occurrence
ARDC	advanced reactor design criteria
ASME	American Society of Mechanical Engineers
BDBE	beyond-design-basis event
CFR	*Code of Federal Regulations*
DOE	U.S. Department of Energy
DRACS	direct reactor auxiliary cooling system
EAB	exclusion area boundary
ECCS	emergency core cooling system
FAUNA	Forschungsanlage zur Untersuchung nuklearer Aerosole (Research Facility for Investigating Nuclear Aerosols)
FHR	fluoride high-temperature reactors
GCR	gas-cooled fast reactors
GDC	general design criterion
HTGR	high-temperature gas-cooled reactor
IAEA	International Atomic Energy Agency
LOCA	loss of coolant accident
LFR	lead fast reactor
LMR	liquid-metal reactor
LPZ	low-population zone
LWR	light-water reactor
MHTGR	modular high-temperature gas-cooled reactor
MHTGR-DC	MHTGR design criteria
MSR	molten salt reactors
NaK	sodium-potassium
NGNP	Next Generation Nuclear Plant
NRC	U.S. Nuclear Regulatory Commission
OMB	Office of Management and Budget
PDC	principal design criteria
PRISM	Power Reactor Innovative Small Module
RCCS	reactor cavity cooling system
RCPB	reactor coolant pressure boundary
RG	regulatory guide
SAFR	Sodium Advanced Fast Reactor
SARRDL	specified acceptable system radionuclide release design limit
SFR	sodium-cooled fast reactor
SFR-DC	SFR design criteria
SRM	staff requirements memorandum
SSC	structure, system, and component

TRISO	tristructural isotropic fuel

REFERENCES[7]

1. *U.S. Code of Federal Regulations*, "Domestic Licensing of Production and Utilization Facilities," Part 50, Chapter 1, Title 10, "Energy." (10 CFR Part 50)

2. *U.S. Code of Federal Regulations*, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Part 52, Title 10, "Energy." (10 CFR Part 52)

3. U.S. Nuclear Regulatory Commission, NUREG-1338, "Draft Preapplication Safety Evaluation Report for the Modular High-Temperature Gas-Cooled Reactor (MHTGR)," December 1995. (Agencywide Documents Access and Management System (ADAMS) Accession No. ML052780497).

4. U.S. Nuclear Regulatory Commission, NUREG-1368, "Preapplication Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Liquid-Metal Reactor," February 1994. (ADAMS Accession No. ML063410561).

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9. U.S. Nuclear Regulatory Commission, SECY-03-0047, "Policy Issues Related to Licensing Non-Light-Water Reactor Designs," March 2003. (ADAMS Accession No. ML030160002).

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11. U.S. Nuclear Regulatory Commission, "Next Generation Nuclear Plant — Assessment of Key Licensing Issues," July 17, 2014. (ADAMS Accession Nos. ML14174A734, ML14174A774 (Enclosure 1), and ML14174A845 (Enclosure 2)).

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7 Publicly available NRC published documents are available electronically through the NRC Library on the NRC's public Web site at http://www.nrc.gov/reading-rm/doc-collections/ and through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html. The documents can also be viewed online or printed for a fee in the NRC's Public Document Room (PDR) at 11555 Rockville Pike, Rockville, MD. For problems with ADAMS, contact the PDR staff at 301-415-4737 or (800) 397-4209; fax (301) 415-3548; or e-mail pdr.resource@nrc.gov.

12. U.S. Nuclear Regulatory Commission, "Policy Statement on the Regulation of Advanced Reactors" (73 FR 60612), October 14, 2008. (ADAMS Accession No. ML082750370).

13. U.S. Nuclear Regulatory Commission, "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants," August 1985. (ADAMS Accession No. ML003711521).

14. U.S. Nuclear Regulatory Commission, SECY-15-0002, "Proposed Updates of Licensing Policies Rules, and Guidance for Future New Reactor Applications," January 2015. (ADAMS Accession Nos. ML13281A382, ML13277A647 (Enclosure 1), ML13277A652 (Enclosure 2)).

15. U.S. Nuclear Regulatory Commission, SRM-SECY-15-002, "Staff Requirements – SECY-15-0002 – Proposed Updates of Licensing Policies, Rules, and Guidance for Future New Reactor Applications" September 22, 2015, (ADAMS Accession No. ML15266A023).

16. U.S. Nuclear Regulatory Commission, "NRC Vision and Strategy: Safely Achieving Effective and Efficient Non-Light Water Reactor Mission Readiness," December 2016. (ADAMS Accession No. ML16356A670).

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8 Copies of U.S. Department of Energy (DOE) documents may be obtained from DOE at 1000 Independence Avenue, SW, Washington DC, 20585 or electronically from their web site: www.doe.gov.

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10 Copies of International Atomic Energy Agency (IAEA) documents may be obtained through their Web site: WWW.IAEA.Org/ or by writing the International Atomic Energy Agency P.O. Box 100 Wagramer Strasse 5, A-1400 Vienna, Austria. Telephone (+431) 2600-0, Fax (+431) 2600-7, or E-Mail at Official.Mail@IAEA.Org.

11 Copies of International Atomic Energy Agency (IAEA) documents may be obtained through their Web site: WWW.IAEA.Org/ or by writing the International Atomic Energy Agency P.O. Box 100 Wagramer Strasse 5, A-1400 Vienna, Austria. Telephone (+431) 2600-0, Fax (+431) 2600-7, or E-Mail at Official.Mail@IAEA.Org.

12 Copies of Annals of Nuclear Energy Articles may be obtained through the Science Direct Web site: WWW.ScienceDirect.com.

APPENDIX A

ADVANCED REACTOR DESIGN CRITERIA

The table below contains the advanced reactor design criteria (ARDC). These criteria are generally applicable to six different types of non-light-water reactor (LWR) technologies (e.g., sodium-cooled fast reactors (SFRs), lead-cooled fast reactors, gas-cooled fast reactors, modular high-temperature gas-cooled reactors (MHTGRs), fluoride high-temperature reactors, and molten salt reactors). Applicants/designers may use the ARDC in this appendix to develop all or part of the principal design criteria (PDC) and may choose among the ARDC, SFR-DC (Appendix B), or MHTGR-DC (Appendix C) to develop each PDC. Applicants/designers may also develop entirely new PDC as needed to address unique design features in their respective designs.

To develop these ARDC, the staff of the U.S. Nuclear Regulatory Commission (NRC) reviewed each general design criterion (GDC) in Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," to determine its applicability to non-LWR designs. The NRC staff then determined what, if any, adaptation was appropriate for non-LWRs. The results are included in Column 2 of the table below. The table also includes the NRC staff's rationale for the adaptations. In many cases, the rationale refers to changes made to the language of the GDC. To fully understand the context of the rationale, the user of this RG should refer to the appropriate GDC. Where the NRC staff determined that the current GDC were applicable to the ARDC, the table denotes "Same as GDC."

The results of this review are presented in the table below, which has three columns:

Column 1—Criterion Number
Column 2—ARDC Title and Content
Column 3—NRC Rationale for Adaptations to GDC

The table is further divided into six sections similar to 10 CFR Part 50, Appendix A:

Section I—Overall Requirements (Criterion 1 – 15)
Section II — Multiple Barriers (Criterion 10 – 20)
Section III — Reactivity Control (Criterion 21 – 29)
Section IV — Fluid Systems (Criterion 30 – 46)
Section V — Reactor Containment (Criterion 50 – 57)
Section VI — Fuel and Radioactivity Control (Criterion 60 – 64)

APPENDIX A. ADVANCED REACTOR DESIGN CRITERIA

I. Overall Requirements		
Criterion	**ARDC Title and Content**	**NRC Rationale for Adaptions to GDC**
1	*Quality standards and records.* Same as GDC 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.	
2	*Design bases for protection against natural phenomena.* Same as GDC 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.	

APPENDIX A. ADVANCED REACTOR DESIGN CRITERIA

I. Overall Requirements		
Criterion	**ARDC Title and Content**	**NRC Rationale for Adaptions to GDC**
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 fire- resistant materials shall be used wherever practical throughout the unit, particularly in locations with structures, systems, 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 ensure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.	The phrase containing examples where noncombustible and heat-resistant materials must be used has been broadened to apply to all advanced reactor designs. Instead of "and," the phrase "locations with structures, systems, and components (SSCs) important to safety" uses "or," which is logically correct in this case.
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. 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.	This change removes the light-water reactor (LWR) emphasis on loss-of-coolant accidents (LOCAs) that may not apply to every design. 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 ensure that fuel integrity is maintained. Therefore, a specific reference to LOCAs is not applicable to all designs. LOCAs may still require analysis in conjunction with postulated accidents if relevant to the design. Reference to pipe whip may not be applicable to designs that operate at low pressure.
5	*Sharing of structures, systems, and components.* Same as GDC	

APPENDIX A. ADVANCED REACTOR DESIGN CRITERIA

I. Overall Requirements		
Criterion	**ARDC Title and Content**	**NRC Rationale for Adaptions to GDC**
	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.	

II. Multiple Barriers		
Criterion	**ARDC Title and Content**	**NRC Rationale for Adaptions to GDC**
10	*Reactor design.* Same as GDC 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.	
11	*Reactor inherent protection.* The reactor core and associated 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.	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.
12	*Suppression of reactor power oscillations.* The reactor core; associated structures; and associated coolant, control, and protection systems shall be designed to ensure that power oscillations that can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.	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.

II. Multiple Barriers		
Criterion	**ARDC Title and Content**	**NRC Rationale for Adaptions to GDC**
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 ensure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.	"Reactor coolant pressure boundary" has been relabeled as "reactor coolant boundary" to create a more broadly applicable non-LWR term that defines the boundary without giving any implication of system operating pressure. As such, the term "reactor coolant boundary" is applicable to non-LWRs that operate at either low or high pressure.
14	*Reactor coolant boundary.* The reactor coolant 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" 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.
15	*Reactor coolant system design.* The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to ensure that the design conditions of the reactor coolant boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.	"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.
16	*Containment design.* Same as GDC 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.	For non-LWR technologies other than SFRs and MHTGRs, designers may use the current GDC to develop applicable principal design criteria. The assumed degree of leak tightness for a containment is used within safety analyses and plant performance requirements to confirm onsite and offsite doses are below limits as specified in 10 CFR 50.34. It is also recognized that characteristics of the coolants, fuels, and containments to be used in non-LWR designs could share common features with SFRs and MHTGRs. Hence, designers may propose using the SFR-DC-16 or

II. Multiple Barriers

Criterion	ARDC Title and Content	NRC Rationale for Adaptions to GDC
		MHTGR-DC 16 as appropriate. Use of the MHTGR-DC 16 will be subject to a policy decision by the Commission.
17	*Electric power systems.* Electric power systems shall be provided when required to permit functioning of structures, systems, and components. The safety function for each power system shall be to provide sufficient capacity and capability to ensure that (1) that the design limits for the fission product barriers are not exceeded as a result of anticipated operational occurrences and (2) safety functions that rely on electric power are maintained in the event of postulated accidents. The electric power systems shall include an onsite power system and an additional power system. The onsite electric power system shall have sufficient independence, redundancy, and testability to perform its safety functions, assuming a single failure. An additional power system shall have sufficient independence and testability to perform its safety function. If electric power is not needed for anticipated operational occurrences or postulated accidents, the design shall demonstrate that power for important to safety functions is provided.	The electric power systems are required to provide reliable power for SSCs during anticipated operational occurrences or postulated accident conditions when those SSCs' safety functions require electric power. The safety functions are established by the safety analyses (i.e. design basis accidents). Where electric power is needed for anticipated operational occurrences or postulated accidents, the electric power systems shall be sufficient in capacity and capability to ensure that safety functions as well as important to safety functions are maintained. The electric power systems provide redundancy and defense-in-depth since there would be a minimum of two power systems. Compared to GDC 17, more emphasis is placed herein on requiring reliability of the overall power supply scheme rather than fully prescribing how such reliability can be attained. For example, reference to offsite electric power systems was deleted to provide for those reactor designs that do not depend on offsite power for the functioning of SSCs important to safety or do not connect to a power grid. The onsite power system is envisioned as a fully Class 1E power system and the additional power system is left to the discretion of the designer as long as it meets the performance criteria in paragraph one and the design criteria of paragraph two. For example, the additional independent power source could be from the electrical grid, a diesel generator, a combustion gas turbine or some other alternative, again, at the discretion of the designer. In this context, important to safety functions refer to the broader, potentially non-safety related functions such as post-accident

II. Multiple Barriers		
Criterion	**ARDC Title and Content**	**NRC Rationale for Adaptions to GDC**
		monitoring, control room habitability, emergency lighting, radiation monitoring, communications and/or any others that may be deemed appropriate for the given design. The electric power system for important to safety functions could be non-Class 1E and would not be required to have redundant power sources.
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 systems.	ARDC 18 is a design-independent companion criterion to ARDC 17. Wording pertaining to additional system examples has been deleted to allow increased flexibility associated with various designs. Specifically, the text related to the nuclear power unit, offsite power system, and onsite power system was deleted to be consistent with ARDC 17.
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. 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 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	ARDC 19 preserves the language of GDC 19 which states (with emphasis added) “A control room shall be provided from which *actions can be taken* to operate the nuclear power unit *safely*…” However some clarification of this language is warranted. It is clear from this language that there is a need to for operators to be able to take “actions” to control the plant. Therefore, designers must consider how the design of *controls* support safe operator actions. In addition, NRC staff recognizes that in order for operators to act “safely” as stated in ARDC, that operators must *have certain knowledge* about the status of the plant and be able to *make decisions* about the appropriate course of action given a particular operating circumstance. Therefore, these cognitive needs

II. Multiple Barriers		
Criterion	**ARDC Title and Content**	**NRC Rationale for Adaptions to GDC**
	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.	of operators should also be considered by designers when interpreting ARDC 19. This consideration should be reflected in the design of indications, displays, alarms, controls or other future technologies which are used to inform operators of plant status and may be used to support the decision making process (such as computer based procedures). This position is consistent with 10 CFR 50.34(f)(2)(iii) which describes the contents required in applications for construction permits. Amongst many other requirements, this rule indicates that the control room design must reflect "state-of-the-art human factors principles." These state-of-the-art principles inherently consider both the cognitive and physical aspects of operator action as described above. The criterion was updated to remove specific emphasis on LOCAs, which may be not appropriate. Reference to "whole body, or its equivalent to any part of the body" has been updated to the current total effective dose equivalent standard as defined in § 50.2. An additional control room habitability requirement has been proposed. It addresses a new concern: accidents that are not radiological in nature 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.

III. Reactivity Control		
Criterion	**ARDC Title and Content**	**NRC Rationale for Adaptions to GDC**
20	*Protection system functions.* Same as GDC 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.	
21	*Protection system reliability and testability.* Same as GDC 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.	
22	*Protection system independence.* Same as GDC 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	

III. Reactivity Control		
Criterion	**ARDC Title and Content**	**NRC Rationale for Adaptions to GDC**
	diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.	
23	*Protection system failure modes.* Same as GDC 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.	
24	*Separation of protection and control systems.* Same as GDC 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.	
25	*Protection system requirements for reactivity control malfunctions.* The protection system shall be designed to ensure that specified acceptable fuel design limits are not exceeded during any anticipated operational occurrence accounting for a single malfunction of the reactivity control systems.	Text has been added to clarify that the protection system is designed to protect the specified acceptable fuel design limits for anticipated operational occurrences (AOOs) in combination with a single failure; the protection system does not have to protect the specified acceptable fuel design limits during a postulated accident in combination with a single failure. The example was deleted to make the ARDC technology inclusive.

III. Reactivity Control		
Criterion	**ARDC Title and Content**	**NRC Rationale for Adaptions to GDC**
26	*Reactivity control systems.* A minimum of two reactivity control systems or means shall provide: (1) A means of inserting negative reactivity at a sufficient rate and amount to assure, with appropriate margin for malfunctions, that the design limits for the fission product barriers are not exceeded and safe shutdown is achieved and maintained during normal operation, including anticipated operational occurrences. (2) A means which is independent and diverse from the other(s), shall be capable of controlling the rate of reactivity changes resulting from planned, normal power changes to assure that the design limits for the fission product barriers are not exceeded. (3) A means of inserting negative reactivity at a sufficient rate and amount to assure, with appropriate margin for malfunctions, that the capability to cool the core is maintained and a means of shutting down the reactor and maintaining, at a minimum, a safe shutdown condition following a postulated accident. (4) A means for holding the reactor shutdown under conditions which allow for interventions such as fuel loading, inspection and repair shall be provided.	Recent licensing activity, associated with the application of GDC 26 and GDC 27 to new reactor designs (ADAMS Accession Nos. ML16116A083 (Ref. 29) and ML16292A589) (Ref. 30), revealed that additional clarity could be provided in the area of reactivity control requirements. ARDC 26 combines the scope of GDC 26 and GDC 27. The development of ARDC 26 is informed by the proposed general design criteria of 1965 (AEC-R 2/49, November 5), 1967 (32 FR 10216) (Ref. 31), current GDC 26 and 27, the definition of safety-related SSC in 10 CFR 50.2, SECY-94-084, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs” (Ref. 32), and the prior application of reactivity control requirements. (1) Currently the second sentence of GDC 26 states, that one of the reactivity control systems shall use control rods and shall be capable of reliably controlling reactivity changes to ensure that, under conditions of normal operation, including AOOs, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The staff recognizes that specifying control rods may not be suitable for advanced reactors. Additionally, reliably controlling reactivity, as applied to GDC 26, has been interpreted as ensuring the control rods are capable of rapidly (i.e., within a few seconds) shutting down the reactor (ADAMS Accession No. ML16292A589) (Ref. 30). The staff changed control rods to “means” in recognition that advanced reactor designs may not rely on control rods to rapidly shut down the reactor (e.g., alternative system designs or inherent feedback mechanisms may be relied upon to perform this function). The wording of “reliably controlling reactivity” in GDC 26 has been replaced with “inserting negative reactivity at a sufficient rate and amount” to more clearly define the requirement. For a non-LWR design the rate of negative reactivity insertion may not

III. Reactivity Control		
Criterion	**ARDC Title and Content**	**NRC Rationale for Adaptions to GDC**
		necessitate rapid (within seconds) insertion but should occur in a time frame such that the fission product barrier design limits are not exceeded. The term "specified acceptable fuel design limits" is replaced with "design limits for fission product barriers" to be consistent with the AOO acceptance criteria while also addressing liquid fueled reactors which may not have SAFDLs. ARDC 10 and ARDC 15 provide the appropriate design limits for the fuel and reactor coolant boundary, respectively. The wording "safe shutdown is achieved and maintained…" has been added again to more clearly define the requirements associated with reliably controlling reactivity in GDC 26. SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs" (Ref. 32), describes the characteristics of a safe shutdown condition as reactor subcriticality, decay heat removal, and radioactive materials containment. ARDC 26 (1) clearly defines that reactor shutdown at any time during the transient is the performance requirement. The distinction between during and following the transient is discussed in (2) below. In regards to safety class, the capability to insert negative reactivity at a rate and amount to preserve the fission product barrier(s) and to shut down the reactor during an AOO is identified as a function performed by safety-related SSCs in the 10 CFR 50.2 definition of safety-related SSCs. (2) The first sentence of GDC 26, states that two independent reactivity control systems of different design principles shall be provided. The third sentence of GDC 26, states that the second reactivity control system shall be capable of reliably controlling the

III. Reactivity Control		
Criterion	**ARDC Title and Content**	**NRC Rationale for Adaptions to GDC**
		rate of changes resulting from planned, normal power changes (including xenon burnout) to assure specified acceptable fuel design limits are not exceeded. ARDC 26 (2) is consistent with the current requirements of the second reactivity control system specified in GDC 26. The words “including xenon burnout” have been deleted as this may not be as important for some non-LWR reactor designs. Also, “of different design principles” from the first sentence of GDC 26 has been replaced with “independent and diverse” to clarify the requirement. The reactivity means given by ARDC 26 (2) is a system important to safety but not necessarily safety-related as it does not mitigate an AOO or accident but is used to control planned, normal reactivity changes such that the design limits for the fission product barriers are preserved thereby minimizing challenges to the safety-related reactivity control means or protection system. The term “independent and diverse” indicates no shared systems or components and a design which is different enough such that no common failure modes exist between the system or means in ARDC 26 (2) and safety-related systems in ARDC 26 (1) and (3). (3) Current GDC 27 states that the reactivity control systems shall be designed to have a combined capability 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. Reliably controlling reactivity, as applied to GDC 27 requires that the reactor achieve and maintain a safe, stable condition, including subcriticality, using only safety related equipment with margin for stuck rods (ADAMS Accession No. ML16116A083) (Ref. 29). ARDC 26 (3) is written to clarify that shut down following a postulated accident using safety-related equipment or means is

III. Reactivity Control		
Criterion	**ARDC Title and Content**	**NRC Rationale for Adaptions to GDC**
		required. The term "following a postulated accident" refers to the time when plant parameters are relatively stable, no additional means of mitigation are needed and margins to acceptance criteria are constant or increasing. ARDC 26 allows for a return to power during a postulated accident consistent with the current licensing basis of some existing PWRs if sufficient heat removal capability exists (e.g., PWR main steam line break accident), but ARDC 26 (1) precludes a return to power during an AOO. (4) The fourth sentence of GDC 26 regarding the capability to reach cold shutdown has been generalized in ARDC 26 (4) to refer to activities which are performed at conditions below (less limiting than) those normally associated with safe shutdown. SECY-94-084 (Ref. 32) describes staff positions on obtaining a cold shutdown and explains that the requirement to bring the plant to cold shutdown is driven by the need to inspect and repair a plant following an accident. In regards to safety class, the capability to bring the plant to a cold shutdown is not covered by the definition of safety-related SSCs in 10 CFR 50.2, and most operating pressurized-water reactors have not credited safety-related SSCs to satisfy this requirement of GDC 26. Based on the information provided above, the system credited for holding the reactor subcritical under conditions necessary for activities such as refueling, inspection and repair is identified as an important to safety system.
27	*Combined reactivity control systems capability.* DELETED—Information incorporated into ARDC 26	
28	*Reactivity limits.* The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to ensure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant boundary greater than limited local yielding nor (2) sufficiently	"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.

III. Reactivity Control

Criterion	ARDC Title and Content	NRC Rationale for Adaptions to GDC
	disturb the core, its support structures, or other reactor vessel internals to impair significantly the capability to cool the core.	The list of "postulated reactivity accidents" has been deleted to make the ARDC technology inclusive.
29	*Protection against anticipated operational occurrences.* Same as GDC 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.	

IV. Fluid Systems

Criterion	ARDC Title and Content	NRC Rationale for Adaptions to GDC
30	*Quality of reactor coolant boundary.* Components that are part of the reactor coolant 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.	"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.
31	*Fracture prevention of reactor coolant boundary.* The reactor coolant boundary shall be designed with sufficient margin to ensure 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, service degradation of material properties, creep, fatigue, stress rupture, and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the	"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. Specific examples are added to the ARDC to account for the high design and operating temperatures, coolant composition, contaminants, and reaction products

IV. Fluid Systems		
Criterion	**ARDC Title and Content**	**NRC Rationale for Adaptions to GDC**
	uncertainties in determining (1) material properties, (2) the effects of irradiation and coolant composition, including contaminants and reaction products, on material properties,, (3) residual, steady-state, and transient stresses, and (4) size of flaws.	
32	*Inspection of reactor coolant boundary.* Components that are part of the reactor coolant boundary shall be designed to permit (1) periodic inspection and functional testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor vessel.	“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. The staff modified the LWR GDC by replacing the term “reactor pressure vessel” with “reactor vessel,” which the staff believes is a more generically applicable term. Functional testing is testing that assesses component and system operational readiness such as required in the ASME OM Code as incorporated by reference in 10 CFR 50.55a and in Plant Technical Specifications.
33	*Reactor coolant inventory maintenance.* A system to maintain reactor coolant inventory for protection against small breaks in the reactor coolant boundary shall be provided as necessary to ensure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant inventory loss due to leakage from the reactor coolant boundary and rupture of small piping or other small components that are part of the boundary. The system shall be designed to ensure that the system safety function can be accomplished using the piping, pumps, and valves used to maintain reactor coolant inventory during normal reactor operation.	ARDC 33 was relabeled as “inventory maintenance” to provide more flexibility for advanced reactor designs. The first sentence is modified so that it ends with “...shall be provided as necessary” and is combined with the second sentence “as necessary to ensure…” (without the opening phrase “The system safety function shall be”) to recognize that the inventory control system may be unnecessary for some designs to maintain safety functions that ensure 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

IV. Fluid Systems		
Criterion	**ARDC Title and Content**	**NRC Rationale for Adaptions to GDC**
		system operating pressure. As such, the term "reactor coolant boundary" is applicable to non-LWRs that operate at either low or high pressure. The staff maintained the words "system safety function" of GDC 33 because reactor coolant inventory maintenance may be necessary in some designs to support residual heat removal, which is a safety function. If not required for maintaining residual heat removal capability, the qualifier "as necessary" in the first sentence would apply. For example, if all small breaks or leaks would result in reactor coolant inventory levels such that the 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. The GDC reference to electric power was removed. Refer to ARDC 17 concerning those systems that require electric power.
34	*Residual heat removal.* A system to remove residual heat shall be provided. For normal operations and anticipated operational occurrences, 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 boundary are not exceeded. Suitable redundancy in components and features and suitable interconnections, leak detection, and isolation capabilities shall be provided to ensure that the system safety function can be accomplished, assuming a single failure.	In most advanced reactor designs, a single system (i.e., the residual heat removal system) is provided to perform both the residual heat removal and emergency core cooling functions. In this case, the single system would be designed to meet the requirements of ARDC 34 and ARDC 35. (for more discussion see NUREG-0968 (Ref. 5) and NUREG-1368 (Ref.4)) However, the staff acknowledges that this may not be the case for every advanced reactor design. Therefore, to allow current and future non-LWR designers the flexibility to provide a single system or multiple systems to perform residual heat removal and emergency core cooling, the staff decided to keep the ARDC 34 and ARDC 35 separate in lieu of combining them into a single criterion. The staff's approach to provide two separate criteria is consistent with the approach taken in the LWR GDCs.

IV. Fluid Systems		
Criterion	**ARDC Title and Content**	**NRC Rationale for Adaptions to GDC**
		"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. The second paragraph addresses residual heat removal system redundancy. The GDC reference to electric power was removed. Refer to ARDC 17 concerning those systems that require electric power.
35	*Emergency core cooling system.* A system to assure sufficient core cooling during postulated accidents and to remove residual heat following postulated accidents shall be provided. The system safety function shall be to transfer heat from the reactor core during and following postulated accidents such that fuel and clad damage that could interfere with continued effective core cooling is prevented.	In most advanced reactor designs, a single system (i.e., the residual heat removal system) is provided to perform both the residual heat removal and emergency core cooling functions. In this case, the single system would be designed to meet the requirements of ARDC 34 and ARDC 35. (for more discussion see NUREG-0968 (Ref. 5) and NUREG-1368 (Ref. 4)) However, the staff acknowledges that this may not be the case for every advanced reactor design. Therefore, to allow current and future non-LWR designers the flexibility to provide a single system or multiple systems to perform residual heat removal and emergency core cooling, the staff decided to keep the ARDC 34 and ARDC 35 separate in lieu of combining them into a single criterion. Effective core cooling may include maintaining the primary coolant boundary in a condition necessary for adequate postulated accident heat removal. The staff's approach to provide two separate criteria is consistent with the approach taken in the LWR GDCs. This change removes the light-water reactor emphasis on loss of coolant accidents that may not apply to every design. Loss of

IV. Fluid Systems		
Criterion	**ARDC Title and Content**	**NRC Rationale for Adaptions to GDC**
		coolant accidents may still require analysis in conjunction with postulated accidents if they are relevant to the design. The GDC reference to electric power was removed. Refer to ARDC 17 concerning those systems that require electric power.
36	*Inspection of emergency core cooling system.* A system that provides emergency core cooling shall be designed to permit appropriate periodic inspection of important components to ensure the integrity and capability of the system.	In most advanced reactor designs, a single system (i.e., the residual heat removal system) is provided to perform both the residual heat removal and emergency core cooling functions. In this case, the single system would be designed to meet the requirements of ARDC 34 and ARDC 35. (for more discussion see NUREG-0968 (Ref. 5) and NUREG-1368 (Ref. 4)) However, the staff acknowledges that this may not be the case for every advanced reactor design. Therefore, to allow current and future non-LWR designers the flexibility to provide a single system or multiple systems to perform residual heat removal and emergency core cooling, the staff decided to keep the ARDC 34 and ARDC 35 separate in lieu of combining them into a single criterion. The staff's approach to provide two separate criteria is consistent with the approach taken in the LWR GDCs. The ARDC has slightly different wording than the GDC to clarify the scope of the criterion. Any system, or portions of a system, credited with an emergency core cooling function during postulated accidents (for example, a system that performs both the residual heat removal function and the emergency core cooling function) would need to meet ARDC 36. The list of examples 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-inclusive ARDC.

IV. Fluid Systems		
Criterion	**ARDC Title and Content**	**NRC Rationale for Adaptions to GDC**
		Review of the proposed DOE SFR and MHTGR DC found that only the SFR provided specific examples of important components but were generic in nature and did not include any significant additional guidance.
37	*Testing of emergency core cooling system.* A system that provides emergency core cooling shall be designed to permit appropriate periodic functional testing to ensure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the system components, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of any associated systems and interfaces necessary to transfer decay heat to the ultimate heat sink.	In most advanced reactor designs, a single system (i.e., the residual heat removal system) is provided to perform both the residual heat removal and emergency core cooling functions. In this case, the single system would be designed to meet the requirements of ARDC 34 and ARDC 35. (for more discussion see NUREG-0968 (Ref. 5) and NUREG-1368 (Ref. 4)) However, the staff acknowledges that this may not be the case for every advanced reactor design. Therefore, to allow current and future non-LWR designers the flexibility to provide a single system or multiple systems to perform residual heat removal and emergency core cooling, the staff decided to keep the ARDC 34 and ARDC 35 separate in lieu of combining them into a single criterion. The staff's approach to provide two separate criteria is consistent with the approach taken in the LWR GDCs. The ARDC has slightly different wording than the GDC to clarify the scope of the criterion. Any system, or portions of a system, credited with an emergency core cooling function during postulated accidents (for example, a system that performs both the residual heat removal function and the emergency core cooling function) would need to meet ARDC 37. 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.

IV. Fluid Systems		
Criterion	**ARDC Title and Content**	**NRC Rationale for Adaptions to GDC**
		Functional testing is testing that assesses component and system operational readiness such as required in the ASME OM Code as incorporated by reference in 10 CFR 50.55a and in Plant Technical Specifications. A non-leaktight system may be acceptable for some designs provided that (1) the system leakage does not impact safety functions under all conditions, and (2) defense in depth is not impacted by system leakage. "Active" has been deleted in item (2) as appropriate operability and performance system component testing are required, regardless of an active or passive nature. Reference to the operation of applicable portions of the protection system, structural and equipment cooling system, and power transfers is considered part of the more general "associated systems." Together with the ultimate heat sink, they are part of the operability testing of the system as a whole. The GDC reference to electric power was removed. Refer to ARDC 17 concerning those systems that require electric power.
38	*Containment heat removal.* A system to remove heat from the reactor containment shall be provided as necessary to maintain the containment pressure and temperature within acceptable limits following postulated accidents. Suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to ensure that the system safety function can be accomplished, assuming a single failure.	"…as necessary…" is meant to condition an ARDC 38 application to designs requiring heat removal for conventional containments that are found to require heat removal measures. The 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 the second paragraph.

IV. Fluid Systems		
Criterion	**ARDC Title and Content**	**NRC Rationale for Adaptions to GDC**
39	*Inspection of containment heat removal system.* The containment heat removal system shall be designed to permit appropriate periodic inspection of important components to ensure the integrity and capability of the system.	Examples were deleted to make the ARDC technology inclusive.
40	*Testing of containment heat removal system.* The containment heat removal system shall be designed to permit appropriate periodic functional testing to ensure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the system components, and (3) the operability of the system as a whole, and under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including the operation of associated systems.	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 containment heat removal. Functional testing is testing that assesses component and system operational readiness such as required in the ASME OM Code as incorporated by reference in 10 CFR 50.55a and in Plant Technical Specifications. A non-leaktight system may be acceptable for some designs provided that (1) the system leakage does not impact safety functions under all conditions, and (2) defense in depth is not impacted by system leakage. Reference to the operation of applicable portions of the protection system, structural and equipment cooling system, and power transfers is considered part of the more general "associated systems" for operability testing of the system as a whole. The GDC reference to electric power was removed. Refer to ARDC 17 concerning those systems that require electric power.
41	*Containment atmosphere cleanup.* Systems to control fission products and other substances that 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 other substances in	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" was retained to allow for exceptions.

IV. Fluid Systems		
Criterion	**ARDC Title and Content**	**NRC Rationale for Adaptions to GDC**
	the containment atmosphere following postulated accidents to ensure that containment integrity and other safety functions are maintained. Each system shall have suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capabilities to ensure that its safety function can be accomplished, assuming a single failure.	Considering that a passive containment cooling system may be used or that the containment may have an additional safety function other than radionuclide retention, additional wording for maintaining safety functions is added. The GDC reference to electric power was removed. Refer to ARDC 17 concerning those systems that require electric power.
42	*Inspection of containment atmosphere cleanup systems.* Same as GDC 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.	
43	*Testing of containment atmosphere cleanup systems.* The containment atmosphere cleanup systems shall be designed to permit appropriate periodic functional testing to ensure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the system components, 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 the operation of associated systems.	"Active" has been deleted in item (2), as appropriate operability and performance testing of system components is required regardless of an active or passive nature, as are cited 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 that 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. A non-leaktight system may be acceptable for some designs provided that (1) the system leakage does not impact safety functions under all conditions, and (2) defense in depth is not impacted by system leakage.

IV. Fluid Systems		
Criterion	**ARDC Title and Content**	**NRC Rationale for Adaptions to GDC**
		Functional testing is testing that assesses component and system operational readiness such as required in the ASME OM Code as incorporated by reference in 10 CFR 50.55a and in Plant Technical Specifications. The GDC reference to electric power was removed. Refer to ARDC 17 concerning those systems that require electric power.
44	*Structural and equipment cooling*. A system to transfer heat from structures, systems, and components important to safety to an ultimate heat sink shall be provided, as necessary, 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 ensure that the system safety function can be accomplished, assuming a single failure.	This renamed ARDC accounts for advanced reactor design system differences to include cooling requirements for SSCs, if applicable; this ARDC does not address the residual heat removal system required under ARDC 34, and emergency core cooling system (ECCS) system under ARDC 35 The GDC reference to electric power was removed. Refer to ARDC 17 concerning those systems that require electric power.
45	*Inspection of structural and equipment cooling systems*. The structural and equipment cooling systems shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to ensure the integrity and capability of the systems.	This renamed ARDC accounts for advanced reactor system design differences to include possible cooling requirements for SSCs important to safety.
46	*Testing of structural and equipment cooling systems.* The structural and equipment cooling systems shall be designed to permit appropriate periodic functional testing to ensure (1) the structural and leaktight integrity of their components, (2) the operability and performance of the system components, 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 the systems into operation	This renamed ARDC accounts for advanced reactor system design differences to include possible cooling requirements for SSCs important to safety. 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. A non-leaktight system may be acceptable for some designs provided that (1) the system leakage

IV. Fluid Systems		
Criterion	**ARDC Title and Content**	**NRC Rationale for Adaptions to GDC**
	for reactor shutdown and postulated accidents, including the operation of associated systems.	does not impact safety functions under all conditions, and (2) defense in depth is not impacted by system leakage. Functional testing is testing that assesses component and system operational readiness such as required in the ASME OM Code as incorporated by reference in 10 CFR 50.55a and in Plant Technical Specifications. “Active” has been deleted in item (2) because appropriate operability and performance tests of system components are required regardless of their active or passive nature. The LOCA reference has been removed to provide for any postulated accident that might affect subject SSCs. The GDC reference to electric power was removed. Refer to ARDC 17 concerning those systems that require electric power.

V. Reactor Containment		
Criterion	**ARDC Title and Content**	**NRC Rationale for Adaptions to GDC**
50	*Containment design basis.* The 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. This margin shall reflect consideration of (1) the effects of potential energy sources that have not been included in the determination of the peak conditions, (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.	ARDC 50 specifically addresses a containment structure in the opening sentence and ARDC 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 word "reactor" was removed because the containment is a barrier between the fission products and the environment. There are diverse advanced reactor designs and, hence, there is no single containment concept. 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 non-LWR designs. The example at the end of subpart 1 of the GDC is LWR specific and therefore deleted.
51	*Fracture prevention of containment pressure boundary.* The boundary of the containment structure shall be designed with sufficient margin to ensure that, under operating, maintenance, testing, and postulated accident conditions, (1) its 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.	ARDC 51–57 support ARDC 50, which specifically applies to non-LWR designs that use a fixed containment structure. Therefore, the word "structure" is added to each of these ARDC to clearly convey the understanding that this criterion applies to designs employing containment structures. The word "reactor" was removed because the containment is a barrier between the fission products and the environment. There are diverse advanced reactor designs and, hence, there is no single containment concept. The term "ferritic" was removed to avoid limiting the scope of the criterion to ferritic materials. With this revision, the staff believes

V. Reactor Containment		
Criterion	**ARDC Title and Content**	**NRC Rationale for Adaptions to GDC**
		that this criterion is more broadly applicable to all non-LWR designs. The word “pressure” was left in the title to reflect that, while a design might not have a high-pressure containment like a traditional LWR, the containment still serves a pressure-retaining function.
52	*Capability for containment leakage rate testing*. The containment structure and other equipment that may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.	ARDC 51–57 support ARDC 50, which specifically applies to non-LWR designs that use a fixed containment structure. Therefore, the word “structure” is added to each of these ARDC to clearly convey the understanding that this criterion applies to designs employing containment structures. The word “reactor” was removed because the containment is a barrier between the fission products and the environment. There are diverse advanced reactor designs and, hence, there is no single containment concept.
53	*Provisions for containment testing and inspection.* The 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 that have resilient seals and expansion bellows.	ARDC 51–57 support ARDC 50, which specifically applies to non-LWR designs that use a fixed containment structure. Therefore, the word “structure” is added to each of these ARDC to clearly convey the understanding that this criterion only applies to designs employing containment structures. The word “reactor” was removed because the containment is a barrier between the fission products and the environment. There are diverse advanced reactor designs and, hence, there is no single containment concept.
54	*Piping systems penetrating containment.* Piping systems penetrating the containment structure shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance	ARDC 51–57 support ARDC 50, which specifically applies to non-LWR designs that use a fixed containment structure. Therefore, the word “structure” is added to each of these ARDC to clearly convey the understanding that this ARDC only applies to designs

V. Reactor Containment		
Criterion	**ARDC Title and Content**	**NRC Rationale for Adaptions to GDC**
	capabilities that reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with the capability to verify, by testing, the operational readiness of any isolation valves and associated apparatus periodically and to confirm that valve leakage is within acceptable limits.	employing containment structures. The word "reactor" was removed because the containment is a barrier between the fission products and the environment. There are diverse advanced reactor designs and, hence, there is no single containment concept. In all cases, the rules for containment penetrations to fulfill containment isolation would apply. How this is accomplished should be left to the designer of the particular advanced reactor design, without being too prescriptive as to whether it is a primary or secondary or reactor containment. There may be a need for a containment structure outside the reactor region. For example, in the MSR design, some of the liquid fuel salt is drawn off to a processing system to clean it up and remove fission products before returning it to the reactor. The liquid fuel salt is highly radioactive and would need a containment around the entire system. Alternatively, in an SFR, the guard vessel would be the primary containment and, in the case of the PRISM design, a dome-shaped structure above it that would be the secondary containment. The secondary containment must also meet the containment isolation requirements. The adjustment to the last sentence enhances the clarity of the sentence with respect to the latest terminology used for periodic valve verification and operational readiness. ASME, International Operation and Maintenance of Nuclear Power Plants, Division 1: OM Code: Section IST (ASME OM Code) defines operational readiness as the ability of a component to perform its specified functions. The ASME OM Code is incorporated by reference in the NRC regulations in 10 CFR 50.55a, including the definition of operational readiness for pumps, valves, and dynamic restraints.
55	*Reactor coolant boundary penetrating containment.* Each line that is part of the reactor coolant boundary and that penetrates the containment structure shall be provided with	ARDC 51–57 support ARDC 50, which specifically applies to non-LWR designs that use a fixed containment structure. Therefore, the word "structure" is added to each of these ARDC to clearly convey

V. Reactor Containment		
Criterion	**ARDC Title and Content**	**NRC Rationale for Adaptions to GDC**
	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 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 ensure 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.	the understanding that this ARDC only applies to designs employing containment structures. The word “reactor” was removed because the containment is a barrier between the fission products and the environment. There are diverse advanced reactor designs and, hence, there is no single containment concept. In all cases, the rules for containment penetrations to fulfill containment isolation would apply. How this is accomplished should be left to the designer of the particular advanced reactor design, without being too prescriptive as to whether it is a primary or secondary or reactor containment. There may be a need for a containment structure outside the reactor region. For example, in the MSR design, some of the liquid fuel salt is drawn off to a processing system to clean it up and remove fission products before returning it to the reactor. The liquid fuel salt is highly radioactive and would need a containment around the entire system. Alternatively, in an SFR, the guard vessel would be the primary containment and, in the case of the PRISM design, a dome-shaped structure above it that would be the secondary containment. The secondary containment must also meet the containment isolation requirements. “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.

V. Reactor Containment		
Criterion	**ARDC Title and Content**	**NRC Rationale for Adaptions to GDC**
56	*Containment isolation.* Each line that connects directly to the containment atmosphere and penetrates the 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 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.	ARDC 51–57 support ARDC 50, which specifically applies to non-LWR designs that use a fixed containment structure. Therefore, the word “structure” is added to each of these ARDC to clearly convey the understanding that this criterion only applies to designs employing containment structures. The word “primary” in the title and the text was removed, and the word “reactor” was also removed because the containment is a barrier between the fission products and the environment. There are diverse advanced reactor designs and, hence, there is no single containment concept. In all cases, the rules for containment penetrations to fulfill containment isolation would apply. How this is accomplished should be left to the designer of the particular advanced reactor design, without being too prescriptive as to whether it is a primary or secondary or reactor containment. There may be a need for a containment structure outside the reactor region. For example, in the MSR design, some of the liquid fuel salt is drawn off to a processing system to clean it up and remove fission products before returning it to the reactor. The liquid fuel salt is highly radioactive and would need a containment around the entire system. Alternatively, in an SFR, the guard vessel would be the primary containment and, in the case of the PRISM design, a dome-shaped structure above it that would be the secondary containment. The secondary containment must also meet the containment isolation requirements.
57	*Closed system isolation valves.* Each line that penetrates the containment structure and is neither part of the reactor coolant boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve, 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	ARDC 51–57 support ARDC 50, which specifically applies to non-LWR designs that use a fixed containment structure. Therefore, the word “structure” is added to each of these ARDC to clearly convey the understanding that this criterion only applies to designs employing containment structures. The word “reactor” was removed because the containment is a barrier between the fission products and the environment. There are diverse advanced reactor

V. Reactor Containment		
Criterion	**ARDC Title and Content**	**NRC Rationale for Adaptions to GDC**
	isolation valve, if 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.	designs and, hence, there is no single containment concept. In all cases, the rules for containment penetrations to fulfill containment isolation would apply. How this is accomplished should be left to the designer of the particular advanced reactor design, without being too prescriptive as to whether it is a primary or secondary or reactor containment. There may be a need for a containment structure outside the reactor region. For example, in the MSR design, some of the liquid fuel salt is drawn off to a processing system to clean it up and remove fission products before returning it to the reactor. The liquid fuel salt is highly radioactive and would need a containment around the entire system. Alternatively, in an SFR, the guard vessel would be the primary containment and, in the case of the PRISM design, a dome-shaped structure above it that would be the secondary containment. The secondary containment also has penetrations and needs containment isolation requirements to be fulfilled. “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.

VI. Fuel and Radioactivity Control		
Criterion	**ARDC Title and Content**	**NRC Rationale for Adaptions to GDC**
60	*Control of releases of radioactive materials to the environment.* Same as GDC 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.	
61	*Fuel storage and handling and radioactivity control.* The fuel storage and handling, radioactive waste, and other systems that may contain radioactivity shall be designed to ensure 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 cooling under accident conditions.	The underlying concept of establishing functional requirements for radioactivity control in fuel storage and fuel handling systems is independent of the design of non-LWR advanced reactors. However, some advanced designs may use dry fuel storage that incorporates cooling jackets that can be liquid cooled or air cooled to remove heat. This modification to this GDC allows for both liquid and air cooling of the dry fuel storage containers.
62	*Prevention of criticality in fuel storage and handling.* Same as GDC Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.	

VI. Fuel and Radioactivity Control		
Criterion	**ARDC Title and Content**	**NRC Rationale for Adaptions to GDC**
63	*Monitoring fuel and waste storage.* Same as GDC 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.	
64	*Monitoring radioactivity releases.* Means shall be provided for monitoring the reactor containment atmosphere, effluent discharge paths, and plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.	The phrase “spaces containing components for recirculation of loss-of-coolant accident fluids” was removed to allow for plant designs that do not have LOCA fluids but may have other similar equipment in spaces where radioactivity should be monitored.

APPENDIX B

SODIUM-COOLED FAST REACTOR DESIGN CRITERIA

The table below contains the sodium-cooled fast reactor design criteria (SFR-DC). These criteria are applicable to SFRs of both pool- and loop-type designs.[13] Applicants/designers may use the SFR-DC in this appendix to develop all or part of the principal design criteria (PDC) and may choose among the advanced reactor design criteria (ARDC) (Appendix A), SFR-DC (Appendix B), or modular high-temperature gas-cooled reactor design criteria (MHTGR)-DC (Appendix C) to develop each PDC. Applicants/designers may also develop entirely new PDC as needed to address unique design features in their respective designs.

To develop the SFR-DC, the staff of the U.S. Nuclear Regulatory Commission (NRC) reviewed each general design criterion (GDC) in Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Domestic Licensing of Production and Utilization Facilities," to determine its applicability to SFR designs. The NRC staff then determined what if any adaptation was appropriate for SFRs. The results are included in Column 2 of the table below. The table also includes the NRC staff's rationale for the adaptations. In many cases, the rationale refers to changes made to the language of the GDC. To fully understand the context of the rationale, the user of this RG should refer to the appropriate GDC. Where the NRC staff determined that the current GDC or the ARDC were applicable to the SFR-DC, the table denotes "Same as GDC" or "Same as ARDC," respectively.

The table consists of three columns:

Column 1—Criterion Number
Column 2—SFR-DC Title and Content
Column 3—Staff Rationale for Adaptations to GDC

The table is further divided into seven sections similar to those in 10 CFR Part 50, Appendix A:

Section I—Overall Requirements (Criteria 1–5)
Section II—Multiple Barriers (Criteria 10–19)
Section III—Reactivity Control (Criteria 20–29)
Section IV—Fluid Systems (Criteria 30–46)
Section V—Reactor Containment (Criteria 50–57)
Section VI—Fuel and Radioactivity Control (Criteria 60–64)
Section VII—Additional SFR-DC (Criteria 70–77)

13 The technology-specific design criteria were developed using available design information, previous NRC pre-application reviews of the design types, and more recent industry and DOE national laboratory initiatives in these technology areas (see Reference 17). It is the responsibility of the designer or applicant to provide and justify the PDC for a specific design.

APPENDIX B. SODIUM-COOLED FAST REACTOR DESIGN CRITERIA

I. Overall Requirements		
Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
1	*Quality standards and records.* Same as GDC 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.	
2	*Design bases for protection against natural phenomena.* Same as GDC 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.	

APPENDIX B. SODIUM-COOLED FAST REACTOR DESIGN CRITERIA

I. Overall Requirements		
Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
3	*Fire protection.* Same as ARDC 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 fire-resistant materials shall be used wherever practical throughout the unit, particularly in locations with structures, systems, 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 ensure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.	The phrase containing examples where noncombustible and fire-resistant materials must be used has been broadened to apply to all advanced reactor designs. Instead of “and,” the phrase “locations with structures, systems, and components (SSCs) important to safety” uses “or,” which is logically correct in this case.
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, anticipated operational occurrences, and postulated accidents, including the effects of liquid sodium and its aerosols and oxidation products. 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.	This change removes the light-water reactor (LWR) emphasis on loss-of-coolant accidents (LOCAs) that may not apply to every design. 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 for LWR designs to ensure that fuel integrity is maintained. Therefore, a specific reference to LOCAs is not applicable to all designs. LOCAs may still require analysis in conjunction with postulated accidents if relevant to the design. The phrase “the environmental conditions associated with anticipated operational occurrences” has been added to ensure that the criterion would apply to all SFR design-basis events, as suggested in NUREG-1368. A new sentence is added to ensure the designer considers the effects of sodium leakage and associated chemical reactions with SSCs important to safety, which must be protected.

APPENDIX B. SODIUM-COOLED FAST REACTOR DESIGN CRITERIA

I. Overall Requirements		
Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
	Chemical consequences of accidents, such as sodium leakage, shall be appropriately considered for the design of structures, systems, and components important to safety, which must be protected.	
5	*Sharing of structures, systems, and components.* Same as GDC 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.	

II. Multiple Barriers		
Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
10	*Reactor design.* Same as GDC 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.	
11	*Reactor inherent protection.* Same as ARDC The reactor core and associated systems that contribute to reactivity feedback shall be designed so that, in the power operating range, the net effect of the prompt inherent nuclear	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.

II. Multiple Barriers		
Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
	feedback characteristics tends to compensate for a rapid increase in reactivity.	
12	*Suppression of reactor power oscillations.* Same as ARDC The reactor core; associated structures; and associated coolant, control, and protection systems shall be designed to ensure that power oscillations that can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.	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.
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 ensure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the primary coolant boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.	"Reactor coolant pressure boundary" has been relabeled as "primary coolant boundary" to conform to standard terms used in the liquid-metal reactor (LMR) industry. The use of the term "primary" indicates that the SFR-DC are applicable to the primary cooling system, not the intermediate cooling system.
14	*Primary coolant boundary.* The primary coolant 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" (RCPB) 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 are 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) (Ref. 4).

II. Multiple Barriers		
Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
15	*Primary coolant system design.* The primary coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to ensure that the design conditions of the primary coolant boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.	"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 are 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) (Ref. 4).
16	*Containment design.* A reactor containment consisting of a low-leakage, pressure-retaining structure surrounding the reactor and its primary cooling system shall be provided to control the release of radioactivity to the environment and to ensure 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 50.34 for postulated accidents.	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 in SECY-93-092 (Ref. 7). Therefore, the Commission agreed that the containment leakage for advanced reactors, similar to and including PRISM, NUREG-1368 (Ref. 4) should not be required to meet the "essentially leaktight" statement in GDC 16. Furthermore, all past, and current, SFR designs use a 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. Reactions of sodium with air or water, sodium fires, and hypothetical reactivity accidents caused by sodium voiding or boiling could release significant energy inside the reactor containment structure. Therefore, a low-leakage, pressure-retaining structure surrounding the reactor and its primary cooling system is required. Note that a design could have a low design pressure for the containment.

II. Multiple Barriers		
Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
		Several technical reports and presentations support the need for a pressure-retaining structure surrounding SFRs. The report, "Experimental Facilities for Sodium Fast Reactor Safety Studies, Task Group on Advanced Reactors Experimental Facilities (TAREF)(Ref. 33), indicates that it is necessary for structures to withstand the thermo-mechanical load caused by sodium fire to avoid fire propagation and dispersion of aerosols. The report, "Safety Design Criteria for GEN IV Sodium-Cooled Fast Reactor Systems," (Ref. 34) notes that the design basis for containment shall consider pressure increase and thermal loads due to sodium fire. During the presentation, "SFR Technology Overview," IAEA Education and Training Seminar on Fast Reactor Science and Technology (Ref. 35), the technical expert noted that low design pressure for the containment basis is the heat produced by a potential sodium fire. In the Annals of Nuclear Energy, the article, "NAFCON-SF: A sodium spray fire code for evaluating thermal consequences in SFR containment," (Ref. 36) notes that Beschreibung der Forschungsanlage zur Untersuchung nuklearer Aerosole (FAUNA) spray fire experiments show peak pressures in containment over 3.5 bar within the first 5 seconds, gradually tapering downwards to less than 3.5 bar at 25 seconds.
17	*Electric power systems.* Same as ARDC Electric power systems shall be provided when required to permit functioning of structures, systems, and components. The safety function for each power system shall be to provide	The electric power systems are required to provide reliable power for SSCs during anticipated operational occurrences or postulated accident conditions when those SSCs' safety functions require electric power. The safety functions are established by the safety analyses (i.e. design basis accidents). Where electric power is

II. Multiple Barriers		
Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
	sufficient capacity and capability to ensure that (1) that the design limits for the fission product barriers are not exceeded as a result of anticipated operational occurrences and (2) safety functions that rely on electric power are maintained in the event of postulated accidents. The electric power systems shall include an onsite power system and an additional power system. The onsite electric power system shall have sufficient independence, redundancy, and testability to perform its safety functions, assuming a single failure. An additional power system shall have sufficient independence and testability to perform its safety function. If electric power is not needed for anticipated operational occurrences or postulated accidents, the design shall demonstrate that power for important to safety functions is provided.	needed for anticipated operational occurrences or postulated accidents, the electric power systems shall be sufficient in capacity and capability to ensure that safety functions as well as important to safety functions are maintained. The electric power systems provide redundancy and defense-in-depth since there would be a minimum of two power systems. Compared to GDC 17, more emphasis is placed herein on requiring reliability of the overall power supply scheme rather than fully prescribing how such reliability can be attained. For example, reference to offsite electric power systems was deleted to provide for those reactor designs that do not depend on offsite power for the functioning of SSCs important to safety or do not connect to a power grid. The onsite power system is envisioned as a fully Class 1E power system and the additional power system is left to the discretion of the designer as long as it meets the performance criteria in paragraph one and the design criteria of paragraph two. For example, the additional independent power source could be from the electrical grid, a diesel generator, a combustion gas turbine or some other alternative, again, at the discretion of the designer. In this context, important to safety functions refer to the broader, potentially non-safety related functions such as post-accident monitoring, control room habitability, emergency lighting, radiation monitoring, communications and/or any others that may be deemed appropriate for the given design. The electric power system for important to safety functions could be non-Class 1E and would not be required to have redundant power sources.
18	*Inspection and testing of electric power systems.* *Same as ARDC.*	ARDC 18 is a design-independent companion criterion to ARDC 17.

II. Multiple Barriers		
Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
	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 systems.	Wording pertaining to additional system examples has been deleted to allow increased flexibility associated with various designs. Specifically, the text related to the nuclear power unit, offsite power system, and onsite power system was deleted to be consistent with ARDC 17.
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. 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, 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. Adequate protection against sodium aerosols shall be provided to permit access and occupancy of the control room under accident conditions.	ARDC 19 preserves the language of GDC 19 which states (with emphasis added) "A control room shall be provided from which *actions can be taken* to operate the nuclear power unit *safely*…" However some clarification of this language is warranted. It is clear from this language that there is a need to for operators to be able to take "actions" to control the plant. Therefore, designers must consider how the design of *controls* support safe operator actions. In addition, NRC staff recognize that in order for operators to act "safely" as stated in ARDC, that operators must *have certain knowledge* about the status of the plant and be able to *make decisions* about the appropriate course of action given a particular operating circumstance. Therefore, these cognitive needs of operators should also be considered by designers when interpreting ARDC 19. This consideration should be reflected in the design of indications, displays, alarms, controls or other future technologies which are

II. Multiple Barriers		
Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
	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.	used to inform operators of plant status and may be used to support the decision making process (such as computer based procedures). This position is consistent with 10 CFR 50.34(f)(2)(iii) which describes the contents required in applications for construction permits. Amongst many other requirements, this rule indicates that the control room design must reflect "state-of-the-art human factors principles." These state-of-the-art principles inherently consider both the cognitive and physical aspects of operator action as described above. The criterion was updated to remove specific emphasis on LOCAs, which may be not appropriate for advanced designs. Reference to "whole body, or its equivalent to any part of the body" has been updated to the current total effective dose equivalent standard as defined in § 50.2. An additional control room habitability requirement has been proposed. It addresses a new concern: accidents that are not radiological in nature may also affect control room access and occupancy. This may include accidental sodium leakage and sodium fire, which could release sodium aerosols. The last paragraph of the GDC has been eliminated for the SFR-DC because it is not applicable to future applicants.

III. Reactivity Control		
Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
20	*Protection system functions.* Same as GDC 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.	
21	*Protection system reliability and testability.* Same as GDC 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.	

III. Reactivity Control		
Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
22	*Protection system independence.* Same as GDC 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.	
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, sodium and sodium reaction products, pressure, steam, water, and radiation) are experienced.	In NUREG-1368, Table 3.3 (page 3-21) (Ref. 4), the 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 parentheses in SFR-DC 23.
24	*Separation of protection and control systems.* Same as GDC 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.	

III. Reactivity Control		
Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
25	*Protection system requirements for reactivity control malfunctions.* Same as ARDC The protection system shall be designed to ensure that specified acceptable fuel design limits are not exceeded during any anticipated operational occurrence accounting for a single malfunction of the reactivity control systems.	Text has been added to clarify that the protection system is designed to protect the specified acceptable fuel design limits for AOOs in combination with a single failure; the protection system does not have to protect the specified acceptable fuel design limits during a postulated accident in combination with a single failure. The example was deleted to make the ARDC technology inclusive.
26	*Reactivity control systems.* Same as ARDC A minimum of two reactivity control systems or means shall provide: (1) A means of inserting negative reactivity at a sufficient rate and amount to assure, with appropriate margin for malfunctions, that the design limits for the fission product barriers are not exceeded and safe shutdown is achieved and maintained during normal operation, including anticipated operational occurrences. (2) A means which is independent and diverse from the other(s), shall be capable of controlling the rate of reactivity changes resulting from planned, normal power changes to assure that the design limits for the fission product barriers are not exceeded. (3) A means of inserting negative reactivity at a sufficient rate and amount to assure, with appropriate margin for malfunctions, that the capability to cool the core is maintained and a means of shutting down the reactor and maintaining, at a minimum, a safe shutdown condition following a postulated accident.	Recent licensing activity, associated with the application of GDC 26 and GDC 27 to new reactor designs (ADAMS Accession Nos. ML16116A083 (Ref. 29) and ML16292A589) (Ref. 30), revealed that additional clarity could be provided in the area of reactivity control requirements. ARDC 26 combines the scope of GDC 26 and GDC 27. The development of ARDC 26 is informed by the proposed general design criteria of 1965 (AEC-R 2/49, November 5), 1967 (32 FR 10216) (Ref. 31), current GDC 26 and 27, the definition of safety-related SSC in 10 CFR 50.2, SECY-94-084, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs” (Ref. 32), and the prior application of reactivity control requirements. (1) Currently the second sentence of GDC 26 states, that one of the reactivity control systems shall use control rods and shall be capable of reliably controlling reactivity changes to ensure that, under conditions of normal operation, including AOOs, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The staff recognizes that specifying control rods may not be suitable for advanced reactors. Additionally, reliably controlling reactivity, as applied to GDC 26, has been interpreted as ensuring the control rods are capable of rapidly (i.e., within a few seconds) shutting down the reactor (ADAMS Accession No. ML16292A589) (Ref. 30).

III. Reactivity Control		
Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
	(4) A means for holding the reactor shutdown under conditions which allow for interventions such as fuel loading, inspection and repair shall be provided.	The staff changed control rods to "means" in recognition that advanced reactor designs may not rely on control rods to rapidly shut down the reactor (e.g., alternative system designs or inherent feedback mechanisms may be relied upon to perform this function). The wording of "reliably controlling reactivity" in GDC 26 has been replaced with "inserting negative reactivity at a sufficient rate and amount" to more clearly define the requirement. For a non-LWR design the rate of negative reactivity insertion may not necessitate rapid (within seconds) insertion but should occur in a time frame such that the fission product barrier design limits are not exceeded. The term "specified acceptable fuel design limits" is replaced with "design limits for fission product barriers" to be consistent with the AOO acceptance criteria while also addressing liquid fueled reactors which may not have SAFDLs. ARDC 10 and ARDC 15 provide the appropriate design limits for the fuel and reactor coolant boundary, respectively. The wording "safe shutdown is achieved and maintained…" has been added again to more clearly define the requirements associated with reliably controlling reactivity in GDC 26. SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs" (Ref. 32), describes the characteristics of a safe shutdown condition as reactor subcriticality, decay heat removal, and radioactive materials containment. ARDC 26 (1) clearly defines that reactor shutdown at any time during the transient is the performance requirement. The distinction between during and following the transient is discussed in (2) below. In regards to safety class, the capability to insert negative reactivity at a rate and amount to preserve the fission product barrier(s) and to

III. Reactivity Control		
Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
		shut down the reactor during an AOO is identified as a function performed by safety-related SSCs in the 10 CFR 50.2 definition of safety-related SSCs. (2) The first sentence of GDC 26, states that two independent reactivity control systems of different design principles shall be provided. The third sentence of GDC 26, states that the second reactivity control system shall be capable of reliably controlling the rate of changes resulting from planned, normal power changes (including xenon burnout) to assure specified acceptable fuel design limits are not exceeded. ARDC 26 (2) is consistent with the current requirements of the second reactivity control system specified in GDC 26. The words “including xenon burnout” have been deleted as this may not be as important for some non-LWR reactor designs. Also, “of different design principles” from the first sentence of GDC 26 has been replaced with “independent and diverse” to clarify the requirement. The reactivity means given by ARDC 26 (2) is a system important to safety but not necessarily safety-related as it does not mitigate an AOO or accident but is used to control planned, normal reactivity changes such that the design limits for the fission product barriers are preserved thereby minimizing challenges to the safety-related reactivity control means or protection system. The term “independent and diverse” indicates no shared systems or components and a design which is different enough such that no common failure modes exist between the system or means in ARDC 26 (2) and safety-related systems in ARDC 26 (1) and (3). (3) Current GDC 27 states that the reactivity control systems shall be designed to have a combined capability of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability

III. Reactivity Control		
Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
		to cool the core is maintained. Reliably controlling reactivity, as applied to GDC 27 requires that the reactor achieve and maintain a safe, stable condition, including subcriticality, using only safety related equipment with margin for stuck rods (ADAMS Accession No. ML16116A083) (Ref. 29). ARDC 26 (3) is written to clarify that shut down following a postulated accident using safety-related equipment or means is required. The term "following a postulated accident" refers to the time when plant parameters are relatively stable, no additional means of mitigation are needed and margins to acceptance criteria are constant or increasing. ARDC 26 allows for a return to power during a postulated accident consistent with the current licensing basis of some existing PWRs if sufficient heat removal capability exists (e.g., PWR main steam line break accident), but ARDC 26 (1) precludes a return to power during an AOO. (4) The fourth sentence of GDC 26 regarding the capability to reach cold shutdown has been generalized in ARDC 26 (4) to refer to activities which are performed at conditions below (less limiting than) those normally associated with safe shutdown. SECY-94-084 (Ref. 32) describes staff positions on obtaining a cold shutdown and explains that the requirement to bring the plant to cold shutdown is driven by the need to inspect and repair a plant following an accident. In regards to safety class, the capability to bring the plant to a cold shutdown is not covered by the definition of safety-related SSCs in 10 CFR 50.2, and most operating pressurized-water reactors have not credited safety-related SSCs to satisfy this requirement of GDC 26. Based on the information provided above, the system credited for holding the reactor subcritical under conditions necessary for activities such as refueling, inspection and repair is identified as an important to safety system.

APPENDIX B. SODIUM-COOLED FAST REACTOR DESIGN CRITERIA

III. Reactivity Control

Criterion	SFR-DC Title and Content	NRC Rationale for Adaption to GDC
27	*Combined reactivity control systems capability.* Same as ARDC DELETED—Information incorporated into SFR-DC 26	
28	*Reactivity limits.* The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to ensure that the effects of postulated reactivity accidents can neither (1) result in damage to the primary coolant boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor vessel internals to impair significantly the capability to cool the core.	“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 are applicable to the primary cooling system, not the intermediate cooling system. The list of “postulated reactivity accidents” has been deleted.
29	*Protection against anticipated operational occurrences.* Same as GDC 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.	

IV. Fluid Systems

Criterion	SFR-DC Title and Content	NRC Rationale for Adaption to GDC
30	*Quality of primary coolant boundary.* Components that are part of the primary coolant 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 primary coolant leakage.	“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 are applicable only to the primary cooling system, not the intermediate cooling system.

IV. Fluid Systems		
Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
		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) (Ref. 4).
31	*Fracture prevention of primary coolant boundary.* The primary coolant boundary shall be designed with sufficient margin to ensure 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, service degradation of material properties, creep, fatigue, stress rupture, 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 and coolant composition, including contaminants and reaction products, on material properties, (3) residual, steady–state, and transient stresses, and (4) size of flaws.	"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 are 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) (Ref. 4). Specific examples are added to the SFR-DC to account for the high design and operating temperatures, coolant composition, contaminants, and reaction products
32	*Inspection of primary coolant boundary.* Components that are part of the primary coolant boundary shall be designed to permit (1) periodic inspection and functional testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor vessel.	"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 are 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) (Ref.4).

IV. Fluid Systems		
Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
		The staff modified the LWR GDC by replacing the term "reactor pressure vessel" with "reactor vessel," which the staff believes is a more generically applicable term. Functional testing is testing that assesses component and system operational readiness such as required in the ASME OM Code as incorporated by reference in 10 CFR 50.55a and in Plant Technical Specifications.
33	*Primary coolant inventory maintenance.* A system to maintain primary coolant inventory for protection against small breaks in the primary coolant boundary shall be provided as necessary to ensure that specified acceptable fuel design limits are not exceeded as a result of primary coolant inventory loss due to leakage from the primary coolant boundary and rupture of small piping or other small components that are part of the boundary. The system shall be designed to ensure that the system safety function can be accomplished using the piping, pumps, and valves used to maintain primary coolant inventory during normal reactor operation.	This SFR-DC was retitled as "inventory maintenance" to provide more flexibility for advanced reactor designs. The first sentence is modified so that it ends with "...shall be provided as necessary" and is combined with the second sentence "as necessary to ensure…" (without the opening phrase, "The system safety function shall be") to recognize that the inventory control system may be unnecessary for some designs to maintain safety functions that ensure fuel design limits are not exceeded. "Reactor coolant pressure boundary" has been relabeled as "primary coolant boundary" to conform to standard terms used in the LMR industry. The SFR primary coolant boundary design requirements differ from the traditional LWR requirements. The effects of low-pressure design are acknowledged in NUREG-1368 (page 3-28) (Ref. 4), in the discussion of GDC 4, and on (page 3-30), under GDC 14. The use of the term "primary" indicates that the SFR-DC is applicable to the primary cooling system, not the intermediate cooling system.

IV. Fluid Systems		
Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
		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. The GDC reference to electric power was removed. Refer to SFR-DC 17 concerning those systems that require electric power.
34	*Residual heat removal.* A system to remove residual heat shall be provided. For normal operations and anticipated operational occurrences, 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 primary coolant boundary are not exceeded. Suitable redundancy in components and features and suitable interconnections leak detection, and isolation capabilities, shall be provided to ensure that the system safety function can be accomplished, assuming a single failure.	In most advanced reactor designs the residual heat removal system is designed to meet the requirements of SFR-DC 34 and SFR-DC 35 (for more discussion see NUREG-0968 (Ref. 5) and NUREG-1368 (Ref. 4)). It is anticipated that the residual heat removal system for non-LWRs will have the same regulatory treatment as the current LWR fleet. “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 are applicable to the primary cooling system, not the intermediate cooling system. The second paragraph addresses residual heat removal system redundancy. The discussion related to sodium leakage and required barriers was moved to a new SFR-DC 78. The GDC reference to electric power was removed. Refer to SFR-DC17 concerning those systems that require electric power.

IV. Fluid Systems		
Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
35	*Emergency core cooling.* Same as ARDC A system to assure sufficient core cooling during postulated accidents and to remove residual heat following postulated accidents shall be provided. The system safety function shall be to transfer heat from the reactor core during and following postulated accidents such that fuel and clad damage that could interfere with continued effective core cooling is prevented. Suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to ensure that the system safety function can be accomplished, assuming a single failure.	In most advanced reactor designs, a single system (i.e. the residual heat removal system) is provided to perform both the residual heat removal and emergency core cooling functions. In this case, the single system would be designed to meet the requirements of SFR-DC 34 and SFR-DC 35. (for more discussion see NUREG-0968 (Ref. 5) and NUREG-1368 (Ref. 4)) However, the staff acknowledges that this may not be the case for every advanced reactor design. Therefore, to allow current and future non-LWR designers the flexibility to provide a single system or multiple systems to perform residual heat removal and emergency core cooling, the staff decided to keep the SFR-DC 34 and SFR-DC 35 separate in lieu of combining them into a single criterion. Effective core cooling may include maintaining the primary coolant boundary in a condition necessary for adequate postulated accident heat removal. The staff's approach to provide two separate criteria is consistent with the approach taken in the LWR GDCs. This change removes the light-water reactor emphasis on loss of coolant accidents that may not apply to every design. Loss of coolant accidents may still require analysis in conjunction with postulated accidents if they are relevant to the design. The discussion related to sodium leakage and required barriers was moved to a new SFR-DC 78. The GDC reference to electric power was removed. Refer to SFR-DC17 concerning those systems that require electric power.
36	*Inspection of emergency core cooling system.* Same as ARDC A system that provides emergency core cooling shall be designed to permit appropriate periodic inspection of important components to ensure the integrity and capability of the system.	In most advanced reactor designs, a single system (i.e. the residual heat removal system) is provided to perform both the residual heat removal and emergency core cooling functions. In this case, the single system would be designed to meet the requirements of SFR-DC 34 and SFR-DC 35. (for more discussion see NUREG-0968

IV. Fluid Systems		
Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
		(Ref. 5) and NUREG-1368 (Ref. 4)) However, the staff acknowledges that this may not be the case for every advanced reactor design. Therefore, to allow current and future non-LWR designers the flexibility to provide a single system or multiple systems to perform residual heat removal and emergency core cooling, the staff decided to keep the SFR-DC 34 and SFR-DC 35 separate in lieu of combining them into a single criterion. The staff's approach to provide two separate criteria is consistent with the approach taken in the LWR GDCs. The SFR-DC has slightly different wording than the GDC to clarify the scope of the criteria. Any system, or portions of a system, credited with an emergency core cooling function during postulated accidents (for example, a system that performs both the residual heat removal function and the emergency core cooling function) would need to meet SFR-DC 36. The list of examples 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-inclusive ARDC. Review of the proposed DOE SFR and MHTGR DC found that only SFR provided specific examples of important components but were generic in nature and did not include any significant additional guidance.
37	*Testing of emergency core cooling system.* Same as ARDC A system that provides emergency core cooling shall be designed to permit appropriate periodic functional testing to ensure (1) the structural and leaktight integrity of its	In most advanced reactor designs, a single system (i.e., the residual heat removal system) is provided to perform both the residual heat removal and emergency core cooling functions. In this case, the single system would be designed to meet the requirements of SFR-DC 34 and SFR-DC 35. (for more discussion see NUREG-0968

IV. Fluid Systems		
Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
	components, (2) the operability and performance of the system components, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of any associated systems and interfaces necessary to transfer decay heat to the ultimate heat sink.	(Ref. 5) and NUREG-1368 (Ref. 4)) However, the staff acknowledges that this may not be the case for every advanced reactor design. Therefore, to allow current and future non-LWR designers the flexibility to provide a single system or multiple systems to perform residual heat removal and emergency core cooling, the staff decided to keep the SFR-DC 34 and SFR-DC 35 separate in lieu of combining them into a single criterion. The staff's approach to provide two separate criteria is consistent with the approach taken in the LWR GDCs. The SFR-DC has slightly different wording than the GDC to clarify the scope of the criteria. Any system, or portions of a system, credited with an emergency core cooling function during postulated accidents (for example, a system that performs both the residual heat removal function and the emergency core cooling function) would need to meet SFR-DC 37. 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. Functional testing is testing that assesses component and system operational readiness such as required in the ASME OM Code as incorporated by reference in 10 CFR 50.55a and in Plant Technical Specifications. A non-leaktight system may be acceptable for some designs provided that (1) the system leakage does not impact safety functions under all conditions, and (2) defense in depth is not impacted by system leakage.

IV. Fluid Systems		
Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
		"Active" has been deleted in item (2) as appropriate operability and performance system component testing are required, regardless of an active or passive nature. Reference to the operation of applicable portions of the protection system, structural and equipment cooling system, and power transfers is considered part of the more general "associated systems." Together with the ultimate heat sink, they are part of the operability testing of the system as a whole. The GDC reference to electric power was removed. Refer to SFR-DC17 concerning those systems that require electric power.
38	*Containment heat removal.* Same as ARDC A system to remove heat from the reactor containment shall be provided as necessary to maintain the containment pressure and temperature within acceptable limits following postulated accidents. Suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to ensure that the system safety function can be accomplished, assuming a single failure.	"…as necessary…" is meant to condition an SFR-DC 38 application to designs requiring heat removal for conventional containments that are found to require heat removal measures. The 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 the second paragraph.
39	*Inspection of containment heat removal system.* Same as ARDC The containment heat removal system shall be designed to permit appropriate periodic inspection of important components to ensure the integrity and capability of the system.	Examples were deleted to make the SFR-DC technology inclusive.
40	*Testing of containment heat removal system.* Same as ARDC	Specific mention of "pressure" testing has been removed yet remains a potential requirement should it be necessary as a

IV. Fluid Systems		
Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
	The containment heat removal system shall be designed to permit appropriate periodic functional testing to ensure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the system components, and (3) the operability of the system as a whole, and under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including the operation of associated systems.	component of "…appropriate periodic functional testing..." of containment heat removal. Functional testing is testing that assesses component and system operational readiness such as required in the ASME OM Code as incorporated by reference in 10 CFR 50.55a and in Plant Technical Specifications. A non-leaktight system may be acceptable for some designs provided that (1) the system leakage does not impact safety functions under all conditions, and (2) defense in depth is not impacted by system leakage. Reference to the operation of applicable portions of the protection system, structural and equipment cooling systems, and power transfers is considered part of the more general "associated systems" for operability testing of the system as a whole. The GDC reference to electric power was removed. Refer to SFR-DC17 concerning those systems that require electric power.
41	*Containment atmosphere cleanup.* Same as ARDC Systems to control fission products and other substances that 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 other substances in the containment atmosphere following postulated accidents to ensure that containment integrity and other safety functions are maintained.	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. Considering that a passive containment cooling system may be used or that the containment may have an additional safety function other than radionuclide retention, additional wording for maintaining safety functions is added.

IV. Fluid Systems		
Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
	Each system shall have suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capabilities to ensure that its safety function can be accomplished, assuming a single failure.	
42	*Inspection of containment atmosphere cleanup systems.* Same as GDC 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.	
43	*Testing of containment atmosphere cleanup systems.* Same as ARDC The containment atmosphere cleanup systems shall be designed to permit appropriate periodic functional testing to ensure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the system components, 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 the operation of associated systems.	"Active" has been deleted in item (2), as appropriate operability and performance testing of system components is required regardless of an active or passive nature, as are cited 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 that 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. A non-leaktight system may be acceptable for some designs provided that (1) the system leakage does not impact safety functions under all conditions and (2) defense in depth is not impacted by system leakage. Functional testing is testing that assesses component and system operational readiness such as required in the ASME OM Code as incorporated by reference in 10 CFR 50.55a and in Plant Technical Specifications.

IV. Fluid Systems		
Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
		The GDC reference to electric power was removed. Refer to SFR-DC17 concerning those systems that require electric power.
44	*Structural and equipment cooling.* Same as ARDC A system to transfer heat from structures, systems, and components important to safety to an ultimate heat sink shall be provided, as necessary, 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 ensure that the system safety function can be accomplished, assuming a single failure.	This renamed SFR-DC accounts for advanced reactor system design differences to include cooling requirements for SSCs, if applicable; this SFR-DC does not address the residual heat removal system required under SFR-DC 34, and ECCS system under SFR-DC 35. The GDC reference to electric power was removed. Refer to SFR-DC17 concerning those systems that require electric power.
45	*Inspection of structural and equipment cooling systems.* Same as ARDC The structural and equipment cooling systems shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to ensure the integrity and capability of the systems.	This renamed ARDC accounts for advanced reactor system design differences to include possible cooling requirements for SSCs important to safety.
46	*Testing of structural and equipment cooling systems.* Same as ARDC The structural and equipment cooling systems shall be designed to permit appropriate periodic functional testing to ensure (1) the structural and leaktight integrity of their components, (2) the operability and performance of the system components, 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 the systems into operation for reactor shutdown and postulated accidents, including the operation of associated systems.	This renamed ARDC accounts for advanced reactor system design differences to include possible cooling requirements for SSCs important to safety. 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. A non-leaktight system may be acceptable for some designs provided that (1) the system leakage does not impact safety functions under all conditions and (2) defense in depth is not impacted by system leakage.

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Criterion	SFR-DC Title and Content	NRC Rationale for Adaption to GDC
		Functional testing is testing that assesses component and system operational readiness such as required in the ASME OM Code as incorporated by reference in 10 CFR 50.55a and in Plant Technical Specifications. "Active" has been deleted in item (2) because appropriate operability and performance tests of system components are required regardless of their active or passive nature. The LOCA reference has been removed to provide for any postulated accident that might affect subject SSCs. The GDC reference to electric power was removed. Refer to SFR-DC17 concerning those systems that require electric power.

V. Reactor Containment

Criterion	SFR-DC Title and Content	NRC Rationale for Adaption to GDC
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 postulated accidents. This margin shall reflect consideration of (1) the effects of potential energy sources that have not been included in the determination of the peak conditions, (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	SFR-DC 50 specifically addresses a containment structure in the opening sentence and SFR-DC 51–57 support the containment structure's design basis. Therefore, SFR-DC 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 non-LWR designs. The example at the end of subpart 1 of the GDC is LWR specific and therefore deleted

V. Reactor Containment		
Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
51	*Fracture prevention of containment pressure boundary.* The boundary of the reactor containment structure shall be designed with sufficient margin to ensure that, under operating, maintenance, testing, and postulated accident conditions, (1) its 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.	SFR-DC 51–57 support SFR-DC 50, which specifically applies to non-LWR designs that use a fixed containment structure. Therefore, the word "structure" is added to each of these SFR-DC to clearly convey the understanding that this criterion applies to designs employing containment structures. The term "ferritic" was removed to avoid limiting the scope of the criterion to ferritic materials. With this revision, the staff believes that this criterion is more broadly applicable. The word "pressure" was left in the title to reflect that, while a design might not have a high-pressure containment like a traditional LWR, the containment still serves a pressure-retaining function. Refer to the SFR-DC 16 rationale for additional information related to SFR containment pressure.
52	*Capability for containment leakage rate testing*. The reactor containment structure and other equipment that may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted to demonstrate resistance at containment design pressure.	SFR-DC 51–57 support SFR-DC 50, which specifically applies to non-LWR designs that use a fixed containment structure. Therefore, the word "structure" is added to each of these SFR-DC to clearly convey the understanding that this criterion applies to designs employing containment structures.
53	*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 that have resilient seals and expansion bellows.	SFR-DC 51–57 support SFR-DC 50, which specifically applies to non-LWR designs that use a fixed containment structure. Therefore, the word "structure" is added to each of these SFR-DC to clearly convey the understanding that this criterion only applies to designs employing containment structures.
54	*Piping systems penetrating containment.* Piping systems penetrating the reactor containment structure shall be provided with leak detection, isolation, and containment capabilities that have redundancy, reliability, and	SFR-DC 51–57 support SFR-DC 50, which specifically applies to non-LWR designs that use a fixed containment structure. Therefore, the word "structure" is added to each of these SFR-DC

V. Reactor Containment		
Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
	performance capabilities necessary to perform the containment safety function and that reflect the importance to safety of preventing radioactivity releases from containment through these piping systems. Such piping systems shall be designed with the capability to verify, by testing, the operational readiness of any isolation valves and associated apparatus periodically and to confirm that valve leakage is within acceptable limits.	to clearly convey the understanding that this criterion only applies to designs employing containment structures. 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 and the residual heat removal system 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 the associated need for testing. Otherwise, NUREG-1368 (page 3-51) indicates that GDC 54 is applicable as written. American National Standards Institute/American Nuclear Society (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," for liquid metal reactors. 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 American Society of Mechanical Engineers (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	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
55	*Primary coolant boundary penetrating containment.* Each line that is part of the primary coolant boundary and that penetrates the 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 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 ensure 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	SFR-DCs 51–57 support SFR-DC 50, which specifically applies to advanced non-LWR designs that use a fixed containment structure. Therefore, the word “structure” is added to each of these SFR-DCs to clearly convey the understanding that this criterion only applies to designs employing containment structures. The title of SFR-DC 55 is the *“Primary coolant boundary penetrating containment.”* The SFR intermediate coolant system is a separate closed system that does not allow any direct mixing of intermediate fluid with the primary coolant sodium. The tubing of the intermediate heat exchanger and associated intermediate coolant system piping 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 coolant system, that penetrates containment and is not part of the primary coolant boundary (in its entirety). This is similar to the treatment of the main steam system and the steam generator in a pressurized-water reactor. “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. 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).

V. Reactor Containment		
Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
	consideration of the population density, use characteristics, and physical characteristics of the site environs.	
56	*Containment isolation.* Each line that connects directly to the containment atmosphere and penetrates the 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 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.	SFR-DC 51–57 support SFR-DC 50, which specifically applies to non-LWR designs that use a fixed containment structure. Therefore, the word “structure” is added to each of these SFR-DC to clearly convey the understanding that this criterion only applies to designs employing containment structures. The word “primary” in the title and the text was removed, and the word “reactor” was also removed because the containment is a barrier between the fission products and the environment. There are diverse advanced reactor designs and, hence, there is no single containment concept. In all cases, the rules for containment penetrations to fulfill containment isolation would apply. How this is accomplished should be left to the designer of the particular advanced reactor design, without being too prescriptive as to whether it is a primary or secondary or reactor containment. There may be a need for a containment structure outside the reactor region. For example, in the MSR design, some of the liquid fuel salt is drawn off to a processing system to clean it up and remove fission products before returning it to the reactor. The liquid fuel salt is highly radioactive and would need a containment around the entire system. Alternatively, in an SFR, the guard vessel would be the primary containment and, in the case of the PRISM design, a dome-shaped structure above it that would be the secondary containment. The secondary containment must also meet the containment isolation requirements.
57	*Closed system isolation valves.* Each line that penetrates the reactor containment structure and is neither part of the primary coolant boundary nor connected directly to the containment atmosphere shall have at least one	SFR-DCs 51–57 support SFR-DC 50, which specifically applies to advanced non-LWR designs that use a fixed containment structure. Therefore, the word “structure” is added to each of these SFR-DCs

V. Reactor Containment

Criterion	SFR-DC Title and Content	NRC Rationale for Adaption to GDC
	containment isolation valve 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 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.	to clearly convey the understanding that this criterion only applies to designs employing containment structures. "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. 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).

VI. Fuel and Reactivity Control

Criterion	SFR-DC Title and Content	NRC Rationale for Adaption to GDC
60	*Control of releases of radioactive materials to the environment.* Same as GDC 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.	

VI. Fuel and Reactivity Control		
Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
61	*Fuel storage and handling and radioactivity control.* Same as ARDC The fuel storage and handling, radioactive waste, and other systems that may contain radioactivity shall be designed to ensure 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 cooling under accident conditions.	The underlying concept of establishing functional requirements for radioactivity control in fuel storage and fuel handling systems is independent of the design of non-LWR 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.
62	*Prevention of criticality in fuel storage and handling.* Same as GDC Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.	
63	*Monitoring fuel and waste storage.* Same as GDC 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.	
64	*Monitoring radioactivity releases.* Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for primary system sodium and cover gas cleanup and processing, effluent discharge paths, and the plant environs for radioactivity that	In NUREG-1368, Table 3.3 (page 3-25), the 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).

VI. Fuel and Reactivity Control		
Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
	may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.	The staff added text to identify other SFR plant areas that should also be included to maintain consideration of all 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.

VII. Additional SFR-DC		
Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
70	*Intermediate coolant system.* If an intermediate cooling system is provided, then the intermediate coolant system shall be designed with sufficient margin to assure that (1) the design conditions of the intermediate coolant boundary are not exceeded during normal operations, including anticipated occupational occurrences, and (2) the integrity of the primary coolant boundary is maintained during postulated accidents.	SFR-DC 70, SFR-DC 75, and SFR-DC 76 describe the three functions of the ICS system: (1) to ensure that the ICS does not impact the safety of the primary coolant system, (2) to ensure that radioactivity in the primary coolant system does not transfer into the power conversion system, and (3) to ensure that the ICS is designed to minimize the possibility of a large, uncontrolled release of sodium. SFR-DC 77 provides verification that the ICS system can perform these functions through inspection. NUREG-1368 (Ref. 4) (page 3-57), 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 (Ref. 5) (Criterion 31, “Design of Intermediate Cooling System,” and Criterion 33, “Inspection of Intermediate Cooling System”). The staff revised SFR-DC 70 to focus on the function of the intermediate coolant system, and to use language that is consistent with other design criteria. The discussion related to sodium leakage and required barriers was moved to a new SFR-DC 78.

VII. Additional SFR-DC		
Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
		Assurance that components of the intermediate coolant system are also designed, as necessary, to prevent the transport of radionuclides between the primary coolant system and the energy conversion system is provided by the design criteria proposed for the intermediate coolant boundary (SFR-DC 75, SFR-DC 76, and SFR-DC 77). Examples of intermediate coolant system accidents would include: rupture (including at a location in the steam-sodium generator), loss of flow, overcooling conditions, and undercooling conditions.
71	*Primary coolant and 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, and (4) air or moisture ingress as a result of a leak of cover gas.	The NRC considered 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'" (pages 12-13) (Ref. 19). NUREG 1368 (Ref. 4) (page 3-57), Section 3.2.4.6, suggested the need for a separate criterion for a sodium and cover gas purity control. Also a separate criterion was included in NUREG-0968 (Ref. 5) (Criterion 34, "Reactor and Intermediate Coolant and Cover Gas Purity Control").
72	*Sodium heating systems.* Heating systems shall be provided for systems and components that are important to safety, and that contain or could be required to contain sodium. These heating systems and their controls shall be appropriately designed to ensure 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 NRC considered 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'" (pages 13-15) (Ref. 19). NUREG-1368 (Ref. 4) (page 3-56), 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 (Ref. 5) (Criterion 7, "Sodium Heating Systems").

VII. Additional SFR-DC		
Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
	the temperature control and the relevant corrective measures associated with that line shall be considered important to safety.	The phrase "and the relevant corrective measures" has been added, in case the cover gas line design includes a feature for clearing an obstruction resulting from condensation or plate out of sodium aerosol or vapor.
73	*Sodium leakage detection and reaction prevention and mitigation*. Means to detect and identify sodium leakage as practical and to limit and control the extent of sodium-air and sodium-concrete reactions and to mitigate the effects of fires resulting from these sodium-air and sodium-concrete reactions shall be provided to ensure that the safety functions of structures, systems, and components important to safety are maintained. Systems from which sodium leakage constitutes a significant safety hazard shall include measures for protection, such as inerted enclosures or guard vessels.	The NRC considered 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'" (pages 15–16) (Ref. 19). NUREG-1368 (Ref. 4) (page 3-56), 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 (Ref. 5) (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 avoid contact between sodium and water and to limit the adverse effects of chemical reactions between sodium and water on the capability of any structure, system, or component to perform any of its intended safety functions. If steam-water is used for energy conversion, 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, including mitigation of the effects of any resulting fire involving sodium.	The NRC considered 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'" (pages 16–18) (Ref.19) NUREG-1368 (Ref 4) (page 3-56), 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 (Ref. 5) (Criterion 4, "Protection against Sodium and NaK reactions"). Fire considerations are added for consistency with SFR-DC 73.
75	*Quality of the intermediate coolant boundary.* Components that are part of the intermediate coolant boundary shall be designed, fabricated, erected, and tested to quality	This criterion is similar to GDC 30 in 10 CFR Part 50, Appendix A, and is intended to ensure that, similar to the reactor coolant pressure boundary, the intermediate coolant boundary is designed,

VII. Additional SFR-DC		
Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
	standards commensurate with the importance of the safety functions to be performed.	fabricated, and tested using quality standards and controls sufficient to ensure that failure of the intermediate system would be unlikely. The statement "commensurate with the system's importance to safety" clarifies that the staff expects a graded approach to be used in determining the quality requirements for the ICS. While not directly applicable to non-LWRs, RG 1.26 "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants" (Ref. 39) provides a basis which can be used to develop a graded quality approach for non-LWR systems including the ICS.
76	*Fracture prevention of the intermediate coolant boundary.* The intermediate coolant boundary shall be designed with sufficient margin to ensure 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.	This criterion addresses the need to maintain a sodium-steam generator in a manner that minimizes the potential for system failure. This criterion is similar to GDC 31 in 10 CFR Part 50, Appendix A, and is intended to ensure that, similar to the reactor coolant pressure boundary, the ICS is designed to prevent a rapid and uncontrolled failure. A sudden rupture of the ICS could result in massive sodium-air, -water, or -concrete reactions and would constitute a risk to the safe operation of the plant and challenge the integrated safety of the plant. This criterion should not be interpreted to preclude the use of rupture discs for controlled, sudden evacuation of the ICS inventory into a vessel or system. The second sentence related to required analyses is removed to make the criteria more generic. In this manner, the design considerations may include, but are not limited to, those previously stated in the design criteria.
77	*Inspection of the intermediate coolant boundary.* Components that are part of the intermediate coolant boundary shall be designed to permit (1) periodic inspection and functional testing of important areas and features to assess their structural and leaktight integrity commensurate with the	This criterion is similar to GDC 32 in 10 CFR Part 50, Appendix A, and is intended to ensure that, similar to the reactor coolant pressure boundary, the intermediate coolant boundary is inspected to ensure that the system is maintained to the quality standard defined in SFR-DC 75.

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Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
	system's importance to safety, and (2) an appropriate material surveillance program for the intermediate coolant boundary.	A non-leaktight system may be acceptable for some designs provided that (1) the system leakage does not impact safety functions under all conditions, and (2) defense in depth is not impacted by system leakage. The staff added "commensurate with the system's importance to safety." If leakage of the intermediate system constitutes a significant risk to the plant, then the appropriate inspection of the intermediate coolant boundary is necessary to ensure that the structural integrity of the boundary is maintained. The requirement for an appropriate surveillance program is maintained to ensure that such a program is provided, as needed, to ensure that the integrity of the intermediate boundary is maintained. At this time, the staff generally does not expect that the projected fluence on the intermediate boundary will be at levels that would necessitate a materials surveillance program that focuses on the impacts of irradiation embrittlement. However, the staff recognizes that this may not be the case for every design. In addition, a materials surveillance program may be used to monitor the effect of other environmental conditions on the boundary materials.
78	*Primary Coolant System Interfaces* When the primary coolant system interfaces with a structure, system, or component containing fluid that is chemically incompatible with the primary coolant, the interface location shall be designed to ensure that the primary coolant is separated from the chemically incompatible fluid by two redundant, passive barriers. When the primary coolant system interfaces with a structure, system, or component containing fluid that is chemically compatible with the primary coolant, then the	The consequence of leakage between the primary coolant system and a heat removal system (i.e. residual heat removal system, intermediate coolant system) is more significant for primary coolant system (potentially impacting the fuel design limits or integrity of the primary coolant boundary) than it is for the heat removal system (coolant drawdown or introduction of radioactive sodium). Rather than creating two parallel requirements for the two systems, SFR-DC 78 was created to discuss leakage and required barriers as a generic criterion. The criterion allows for double walled steam generators, intermediate coolant systems connected to steam power

VII. Additional SFR-DC		
Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
	interface location may be a single passive barrier provided that the following conditions are met: (1) postulated leakage at the interface location does not result in failure of the intended safety functions of structures, systems or components important to safety or result in exceeding the fuel design limits (2) the fluid contained in the structure, system, or component is maintained at a higher pressure than the primary coolant during normal operation, anticipated operational occurrences, shutdown, and accident conditions.	system, and systems similar to the PRISM Direct Reactor Auxiliary Cooling System (DRACS). A paragraph from NUREG 1368 (page 3-41) (Ref. 4) was added describing the characteristics of the residual heat removal working fluid and its associated operating pressure. This SFR-DC has been worded to explain that an intermediate coolant system may be used if the primary coolant is not chemically compatible with the energy conversion system coolant. A single passive barrier is adequate defense in depth when the heat removal working fluid is chemically compatible with the primary coolant, such that postulated leakage between the two systems does not result in the failure of any intended safety function of any SSC important to safety or cause fuel design limits to be exceeded. An example is a heat removal system with liquid sodium potassium (NaK). A liquid sodium primary coolant system that is contaminated with NaK may have phase changes (e.g., solidification, boiling) at different temperatures, without adversely affecting the overall system. The postulated leakage may be based upon a leak-before-break analysis or the ability to detect leakage between the primary and intermediate coolant systems. If the working fluids are not chemically compatible, 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 nonradioactive part of the heat transport system. A sentence has been added to explain that this differential pressure requirement must be satisfied during AOOs and design-basis

VII. Additional SFR-DC		
Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
		accidents, as well as during normal operating and shutdown conditions.
79	*Cover gas inventory maintenance.* A system to maintain cover gas inventory shall be provided as necessary to ensure that the primary coolant sodium design limits are not exceeded as a result of cover gas loss due to leakage from the primary coolant boundary and rupture of small piping or other small components that are part of the primary coolant boundary.	This criterion is similar to GDC 33 in 10 CFR Part 50, Appendix A and SFR-DC 33 in this document. GDC 33 and SFR-DC 33 focus on the effects of primary coolant (sodium) loss. A leak in a SFR primary coolant system may expel the cover gas rather than the primary coolant. The cover gas in the SFR performs an important to safety function by protecting the sodium coolant from chemical reactions. The staff created a new SFR-DC rather than adding the cover gas in the term "primary coolant." The term "primary coolant sodium design limits" is used to maintain consistent terminology with SFR-DC 71. The primary coolant sodium design limits consider the possibility of interactions between the primary coolant sodium and the primary coolant boundary or the fuel due to changes in the chemistry of the primary coolant sodium. The considerations include the possibility of (1) chemical attack, (2) fouling and plugging of passages, (3) radionuclide concentrations, and (4) air or moisture ingress as a result of a leak of cover gas. The term "as necessary" is retained from SFR-DC 33 to permit designer flexibility if leakage of the system does not challenge the design limits of the primary coolant (for instance, an inerted containment filled with Argon). "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 are applicable only to the primary cooling system, not the intermediate cooling system.

VII. Additional SFR-DC		
Criterion	**SFR-DC Title and Content**	**NRC Rationale for Adaption to GDC**
		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) (Ref. 4). The GDC reference to electric power was removed. Refer to SFR-DC 17 concerning those systems that require electric power.

APPENDIX C

MODULAR HIGH-TEMPERATURE GAS-COOLED REACTOR DESIGN CRITERIA

The table below contains the modular high-temperature gas-cooled reactor design criteria (MHTGR-DC).[14] These criteria are applicable to MHTGRs. MHTGR refers to the category of HTGRs that use the inherent high temperature characteristics of tristructural isotropic (TRISO) coated fuel particles, graphite moderator, and helium coolant, as well as passive heat removal from a low power density core with a relatively large height-to-diameter ratio within an uninsulated steel reactor vessel. The MHTGR is designed in such a way to ensure that during design basis events (including loss of forced cooling or loss of helium pressure conditions) radionuclides are retained at their source in the fuel and regulatory requirements for offsite dose are met at the exclusion area boundary. Applicants/designers may use the MHTGR-DC in this appendix to develop all or part of the principal design criteria (PDC) and may choose among the advanced reactor design criteria (ARDC) (Appendix A), sodium-cooled fast reactor design criteria (SFR-DC) (Appendix B), or MHTGR-DC (Appendix C) to develop each PDC. Applicants/designers may also develop entirely new PDC as needed to address unique design features in their respective designs.

To develop these MHTGR-DC, the staff of the U.S. Nuclear Regulatory Commission (NRC) reviewed each general design criterion (GDC) in Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," to determine its applicability to MHTGR designs. The NRC staff then determined what if any adaptation was appropriate for MHTGRs. The results are included in Column 2 of the table below. The table also includes the NRC staff's rationale for the adaptations. In many cases, the rationale refers to changes made to the language of the GDC. To fully understand the context of the rationale, the user of this RG should refer to the appropriate GDC. Where the NRC staff determined that the current GDC or the ARDC were applicable to the MHTGR-DC, the table denotes "Same as GDC," or "Same as ARDC," respectively. In many cases, the NRC staff determined the design criteria were not applicable to MHTGR designs. In these instances, the table denotes "Not applicable to MHTGR."

The table consists of three columns:

Column 1—Criterion Number
Column 2—MHTGR-DC Title and Content
Column 3—Staff Rationale for Adaptations to GDC

The table is further divided into seven sections similar to those in 10 CFR Part 50, Appendix A:

Section I—Overall Requirements (Criteria 1–5)
Section II—Multiple Barriers (Criteria 10–19)
Section III—Reactivity Control (Criteria 20–29)
Section IV—Heat Transport Systems (Criteria 30–46)
Section V—Reactor Containment (Criteria 50–57)

[14] The technology-specific design criteria were developed using available design information, previous NRC pre-application reviews of the design types, and more recent industry and DOE national laboratory initiatives in these technology areas (see Reference 17). It is the responsibility of the designer or applicant to provide and justify the PDC for a specific design.

Section VI—Fuel and Radioactivity Control (Criteria 60–64)
Section VII—Additional MHTGR-DC (Criteria 70–72)

APPENDIX C. MODULAR HIGH-TEMPERATURE GAS-COOLED REACTOR DESIGN CRITERIA

I. Overall Requirements		
Criterion	**MHTGR-DC Title and Content**	**NRC Rationale for Adaptions to GDC**
1	*Quality standards and records.* Same as GDC 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.	
2	*Design bases for protection against natural phenomena.* Same as GDC 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	

I. Overall Requirements		
Criterion	**MHTGR-DC Title and Content**	**NRC Rationale for Adaptions to GDC**
	conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.	
3	*Fire protection.* Same as ARDC 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 fire-resistant materials shall be used wherever practical throughout the unit, particularly in locations with structures, systems, 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 ensure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.	The phrase containing examples where noncombustible and fire-resistant materials must be used has been broadened to apply to all advanced reactor designs. Instead of “and,” the phrase “locations with structures, systems, and components (SSCs) important to safety” uses “or,” which is logically correct in this case.
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. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles originating both inside and outside the reactor helium pressure boundary, 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	This change removes the light-water reactor (LWR) emphasis on loss-of-coolant accidents (LOCAs) that may not apply to every design. 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 ensure that fuel integrity is maintained. Therefore, a specific reference to LOCAs is not applicable to all designs. LOCAs may still require analysis in conjunction with postulated accidents if they are relevant to the design. If an MHTGR design proposes using a direct power cycle in which one or more very high-speed, very high-energy gas turbines are located inside the reactor helium pressure boundary. The presence of one or more very high-energy turbines inside the primary helium pressure boundary creates the potential that a catastrophic

I. Overall Requirements

Criterion	MHTGR-DC Title and Content	NRC Rationale for Adaptions to GDC
	piping rupture is extremely low under conditions consistent with the design basis for the piping.	dynamic failure of the gas turbine (e.g., at power) could result in the consequential catastrophic failure of the primary system pressure boundary caused by the failure of rotating turbine components. To account for the possibility of an MHTGR design that locates high-energy gas turbines inside the reactor helium pressure boundary, the MHTGR-DC language in the area of prevention, protection, and mitigation of turbine dynamic failure is strengthened to support such a power conversion system design approach.
5	*Sharing of structures, systems, and components.* Same as GDC 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.	

II. Multiple Barriers

Criterion	MHTGR-DC Title and Content	NRC Rationale for Adaptions to GDC
10	*Reactor design.* The reactor system and associated heat removal, control, and protection systems shall be designed with appropriate margin to ensure that specified acceptable system radionuclide release design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.	The concept of specified acceptable fuel design limits, which prevent additional fuel failures during anticipated operational occurrences (AOOs), has been replaced with that of the specified acceptable system radionuclide release design limits (SARRDL), which limits the amount of radionuclide inventory that is released by the system under normal and AOO conditions. The term "system" refers to the fuel, the helium coolant circuit and all connected systems that are not isolated and may contribute to dose.

II. Multiple Barriers		
Criterion	**MHTGR-DC Title and Content**	**NRC Rationale for Adaptions to GDC**
		Design features within the reactor system must ensure that the SARRDLS are not exceeded during normal operations and AOOs. The tristructural isotropic (TRISO) fuel used in the MHTGR design is the primary fission product barrier and is expected to have a very low incremental fission product release during AOOs. As noted in NUREG-1338 (Ref. 3) and in the NRC staff's feedback on the Next Generation Nuclear Plant (NGNP) project white paper, "Next-Generation Nuclear Plant – Assessment of Key Licensing Issues" (Ref. 11) the TRISO fuel fission product transport and retention behavior under all expected operating conditions is the key to meeting dose limits, as a different approach to defense in depth is employed in an MHTGR. The SARRDL concept allows for some small increase in circulating radionuclide inventory during an AOO. To ensure the SARRDL is not violated during an AOO, a normal operation radionuclide inventory limit must also be established (i.e., appropriate margin). The radionuclide activity circulating within the helium coolant boundary is continuously monitored such that the normal operation limits and SARRDLs are not exceeded. The SARRDLs 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 NRC has not approved the concept of replacing specified acceptable fuel design limits with SARRDLs. The concept of the TRISO fuel being the primary fission product barrier is intertwined

II. Multiple Barriers		
Criterion	**MHTGR-DC Title and Content**	**NRC Rationale for Adaptions to GDC**
		with 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 "coolant" has been replaced with "heat removal," as helium coolant inventory control for normal operation and AOOs is not necessary to meet the SARRDLs, due to the reactor system design.
11	*Reactor inherent protection.* Same as ARDC The reactor core and associated 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.	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.
12	*Suppression of reactor power oscillations.* The reactor core and associated control and protection systems shall be designed to ensure that power oscillations that can result in conditions exceeding specified acceptable system radionuclide release design limits are not possible or can be reliably and readily detected and suppressed.	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 specified acceptable fuel design limits were replaced by SARRDLs. The discussion on the SARRDL is given in MHTGR-DC 10.
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 ensure adequate safety, including those variables and systems that can affect the fission process and the integrity of the reactor core, reactor helium pressure boundary, and functional containment. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.	"Reactor coolant pressure boundary" has been relabeled as "reactor helium pressure boundary" to conform to standard terms used for MHTGRs. The criterion has been modified to reflect the use of the MHTGR functional containment. See the MHTGR-DC 16 rationale.

II. Multiple Barriers		
Criterion	**MHTGR-DC Title and Content**	**NRC Rationale for Adaptions to GDC**
14	*Reactor helium pressure boundary.* The reactor helium pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, of gross rupture, and of unacceptable ingress of moisture, air, secondary coolant, or other fluids.	“Reactor coolant pressure boundary” has been relabeled as “reactor helium pressure boundary” to conform to standard terms used for MHTGRs. The MHTGR-DC 14 addresses the need to consider leakage of contaminants into the helium used to transport heat from the reactor to the heat exchangers for power production, residual heat removal, and process heat. The phrase “reactor helium pressure boundary” encompasses the entire volume containing helium used to cool the reactor, not just the volume within the reactor vessel. For consistency, a specific requirement is appended to MHTGR-DC 30 for a means of detecting ingress of moisture, air, secondary coolant, or other fluids. Although “other fluids” could be interpreted as including water and steam, for emphasis, the word “moisture” is included in the list of contaminants in both MHTGR-DC 14 and MHTGR-DC 30.
15	*Reactor helium pressure boundary design.* All systems that are part of the reactor helium pressure boundary, such as the reactor system, vessel system, and heat removal systems, and the associated auxiliary, control, and protection systems, shall be designed with sufficient margin to ensure that the design conditions of the reactor helium pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.	“Reactor coolant system” has been relabeled as “reactor helium pressure boundary” to conform to standard terms used for MHTGRs.
16	*Containment design.* A reactor functional containment, consisting of multiple barriers internal and/or external to the reactor and its cooling system, shall be provided to control the release of radioactivity to the environment and to ensure that the functional containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.	The term “functional containment” is applicable to advanced non-LWRs without a pressure retaining containment structure. A functional containment can be defined as “a barrier, or set of barriers taken together, that effectively limit the physical transport and release of radionuclides to the environment across a full range of normal operating conditions, AOOs, and accident conditions.”

II. Multiple Barriers		
Criterion	**MHTGR-DC Title and Content**	**NRC Rationale for Adaptions to GDC**
		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 (e.g., turbine missiles, flooding, aircraft). The MHTGR functional containment safety design objective is to meet 10 CFR 50.34, 52.79, 52.137, or 52.157 offsite dose requirements at the plant's exclusion area boundary (EAB) with margins. The NRC staff has brought the issue of functional containment to the Commission, and the Commission has found it generally acceptable, as indicated in the staff requirements memoranda (SRM) to SECY-93-092 (Ref. 8) and SECY-03-0047 (Ref. 9). In the SRM to SECY-03-0047 (Ref. 10), the Commission instructed the staff to "…develop performance requirements and criteria working closely with industry experts (e.g., designers, EPRI, etc.) and other stakeholders regarding options in this area, taking into account such features as core, fuel, and cooling systems design," and directed the staff to submit options and recommendations to the Commission for a policy decision. The NRC staff also provided feedback to the DOE on this issue as part of the NGNP project. In the NRC staff's "Next Generation Nuclear Plant — Assessment of Key Licensing Issues" (Ref. 11 Enclosure 1), the area on functional containment and fuel development and qualification noted that "…approval of the proposed approach to functional containment for the MHTGR concept, with its emphasis on passive safety features and radionuclide retention within the fuel over a broad spectrum of off-normal conditions, would necessitate that the required fuel particle

II. Multiple Barriers		
Criterion	**MHTGR-DC Title and Content**	**NRC Rationale for Adaptions to GDC**
		performance capabilities be demonstrated with a high degree of certainty." GDC 38, 39, 40, 41, 42, 43, 50, 51, 52, 53, 54, 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 for the performance of the MHTGR reactor building are addressed by new Criterion 71 (design basis) and Criterion 72 (provisions for periodic testing and inspection).
17	*Electric power systems.* Electric power systems shall be provided when required to permit functioning of structures, systems, and components. The safety function for each power system shall be to provide sufficient capacity and capability to ensure that (1) that the specified acceptable system radionuclide release design limits and the reactor helium pressure boundary design limits are not exceeded as a result of anticipated operational occurrences and (2) safety functions that rely on electric power are maintained in the event of postulated accidents. The electric power systems shall include an onsite power system and an additional power system. The onsite electric power system shall have sufficient independence, redundancy, and testability to perform its safety functions, assuming a single failure. An additional power system shall have sufficient independence and testability to perform its safety function. If electric power is not needed for anticipated operational occurrences or postulated accidents, the design shall demonstrate that power for important to safety functions is provided.	The electric power systems are required to provide reliable power for SSCs during anticipated operational occurrences or postulated accident conditions when those SSCs' safety functions require electric power. The safety functions are established by the safety analyses (i.e. design basis accidents). Where electric power is needed for anticipated operational occurrences or postulated accidents, the electric power systems shall be sufficient in capacity and capability to ensure that safety functions as well as important to safety functions are maintained. The electric power systems provide redundancy and defense-in-depth since there would be a minimum of two power systems. Compared to GDC 17, more emphasis is placed herein on requiring reliability of the overall power supply scheme rather than fully prescribing how such reliability can be attained. For example, reference to offsite electric power systems was deleted to provide for those reactor designs that do not depend on offsite power for the functioning of SSCs important to safety or do not connect to a power grid. The onsite power system is envisioned as a fully Class 1E power system and the additional power system is left to the discretion of the designer as long as it meets the performance criteria in

II. Multiple Barriers		
Criterion	**MHTGR-DC Title and Content**	**NRC Rationale for Adaptions to GDC**
		paragraph one and the design criteria of paragraph two. For example, the additional independent power source could be from the electrical grid, a diesel generator, a combustion gas turbine or some other alternative, again, at the discretion of the designer. In this context, important to safety functions refer to the broader, potentially non-safety related functions such as post-accident monitoring, control room habitability, emergency lighting, radiation monitoring, communications and/or any others that may be deemed appropriate for the given design. The electric power system for important to safety functions could be non-Class 1E and would not be required to have redundant power sources. “Reactor coolant pressure boundary” has been relabeled as “reactor helium pressure boundary” to conform to standard terms used for MHTGRs.
18	*Inspection and testing of electric power systems.* Same as ARDC 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 systems.	ARDC 18 is a design-independent companion criterion to ARDC 17. Wording pertaining to additional system examples has been deleted to allow increased flexibility associated with various designs. Specifically, the text related to the nuclear power unit, offsite power system, and onsite power system was deleted to be consistent with ARDC 17.

II. Multiple Barriers		
Criterion	**MHTGR-DC Title and Content**	**NRC Rationale for Adaptions to GDC**
19	*Control room.* Same as ARDC 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. 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 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 condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.	ARDC 19 preserves the language of GDC 19 which states (with emphasis added) "A control room shall be provided from which *actions can be taken* to operate the nuclear power unit *safely*…" However some clarification of this language is warranted. It is clear from this language that there is a need to for operators to be able to take "actions" to control the plant. Therefore, designers must consider how the design of *controls* support safe operator actions. In addition, NRC staff recognize that in order for operators to act "safely" as stated in ARDC, that operators must *have certain knowledge* about the status of the plant and be able to *make decisions* about the appropriate course of action given a particular operating circumstance. Therefore, these cognitive needs of operators should also be considered by designers when interpreting ARDC 19. This consideration should be reflected in the design of indications, displays, alarms, controls or other future technologies which are used to inform operators of plant status and may be used to support the decision making process (such as computer based procedures). This position is consistent with 10 CFR 50.34(f)(2)(iii) which describes the contents required in applications for construction permits. Amongst many other requirements, this rule indicates that the control room design must reflect "state-of-the-art human factors principles." These state-of-the-art principles inherently consider both the cognitive and physical aspects of operator action as described above. The criterion was updated to remove specific emphasis on LOCAs, which may be not appropriate for advanced designs such as the MHTGR.

II. Multiple Barriers		
Criterion	**MHTGR-DC Title and Content**	**NRC Rationale for Adaptions to GDC**
		Reference to “whole body, or its equivalent to any part of the body” has been updated to the current total effective dose equivalent standard as defined in § 50.2. An additional control room habitability requirement has been proposed. It addresses a new concern: accidents that are not radiological in nature may also affect control room access and occupancy. The last paragraph of the GDC has been eliminated for the MHTGR-DC because it is not applicable to future applicants.

III. Reactivity Control		
Criterion	**MHTGR-DC Title and Content**	**NRC Rationale for Adaptions to GDC**
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 ensure that the specified acceptable system radionuclide release design limits 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.	“Specified acceptable fuel design limits” has been replaced with SARRDLs. The concept of using SARRDLs is discussed in MHTGR-DC 10. The quantitative value of the SARRDL will be design specific. The protection aspect of automatic operation, to protect normal operation and AOO limits, to sense accident conditions, and to initiate mitigating equipment has been preserved.
21	*Protection system reliability and testability.* Same as GDC 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	

III. Reactivity Control		
Criterion	**MHTGR-DC Title and Content**	**NRC Rationale for Adaptions to GDC**
	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.	
22	*Protection system independence.* Same as GDC 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.	
23	*Protection system failure modes.* Same as GDC 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.	
24	*Separation of protection and control systems.* Same as GDC	

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Criterion	**MHTGR-DC Title and Content**	**NRC Rationale for Adaptions to GDC**
	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.	
25	*Protection system requirements for reactivity control malfunctions.* The protection system shall be designed to ensure that specified acceptable system radionuclide release design limits are not exceeded during any anticipated operational occurrence, accounting for a single malfunction of the reactivity control systems.	"Specified acceptable fuel design limits" is replaced with SARRDLs. The concept of using SARRDLs is discussed in MHTGR-DC 10.
26	*Reactivity control systems.* A minimum of two reactivity control systems or means shall provide: (1) A means of inserting negative reactivity at a sufficient rate and amount to assure, with appropriate margin for malfunctions, that the specified acceptable system radionuclide release design limits and the reactor helium pressure boundary design limits are not exceeded and safe shutdown is achieved and maintained during normal operation, including anticipated operational occurrences. (2) A means which is independent and diverse from the other(s), shall be capable of controlling the rate of reactivity changes resulting from planned, normal power changes to assure that the specified acceptable system radionuclide release	Recent licensing activity, associated with the application of GDC 26 and GDC 27 to new reactor designs (ADAMS Accession Nos. ML16116A083 (Ref. 29) and ML16292A589) (Ref. 30), revealed that additional clarity could be provided in the area of reactivity control requirements. MHTGR-DC 26 combines the scope of GDC 26 and GDC 27. The development of MHTGR-DC 26 is informed by the proposed general design criteria of 1965 (AEC-R 2/49, November 5), 1967 (32 FR 10216) (Ref. 31), current GDC 26 and 27, the definition of safety-related SSC in 10 CFR 50.2, SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs" (Ref. 32), and the prior application of reactivity control requirements. (1) Currently the second sentence of GDC 26 states, that one of the reactivity control systems shall use control rods and shall be

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Criterion	**MHTGR-DC Title and Content**	**NRC Rationale for Adaptions to GDC**
	design limits and the reactor helium pressure boundary design limits are not exceeded. (3) A means of inserting negative reactivity at a sufficient rate and amount to assure, with appropriate margin for malfunctions, that the capability to cool the core is maintained and a means of shutting down the reactor and maintaining, at a minimum, a safe shutdown condition following a postulated accident. (4) A means for holding the reactor shutdown under conditions which allow for interventions such as fuel loading, inspection and repair shall be provided.	capable of reliably controlling reactivity changes to ensure that, under conditions of normal operation, including AOOs, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The staff recognizes that specifying control rods may not be suitable for advanced reactors. Additionally, reliably controlling reactivity, as applied to GDC 26, has been interpreted as ensuring the control rods are capable of rapidly (i.e., within a few seconds) shutting down the reactor (ADAMS Accession No. ML16292A589) (Ref. 30). The staff changed control rods to “means” in recognition that advanced reactor designs may not rely on control rods to rapidly shut down the reactor (e.g., alternative system designs or inherent feedback mechanisms may be relied upon to perform this function). The wording of “reliably controlling reactivity” in GDC 26 has been replaced with “inserting negative reactivity at a sufficient rate and amount” to more clearly define the requirement. For a non-LWR design the rate of negative reactivity insertion may not necessitate rapid (within seconds) insertion but should occur in a time frame such that the fission product barrier design limits are not exceeded. The term “design limits for fission product barriers” is replaced with “specified acceptable system radionuclide release design limits (SARRDLs)” to be consistent with the AOO acceptance criteria associated with MHTGR-DC 10 (SARRDL) and MHTGR-DC 15 (helium pressure boundary). The wording “safe shutdown is achieved and maintained…” has been added again to more clearly define the requirements associated with reliably controlling reactivity in GDC 26. SECY-94-084, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant

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Criterion	**MHTGR-DC Title and Content**	**NRC Rationale for Adaptions to GDC**
		Designs" (Ref. 32), describes the characteristics of a safe shutdown condition as reactor subcriticality, decay heat removal, and radioactive materials containment. MHTGR-DC 26 (1) clearly defines that reactor shutdown at any time during the transient is the performance requirement. The distinction between during and following the transient is discussed in (2) below. In regards to safety class, the capability to insert negative reactivity at a rate and amount to preserve the fission product barrier(s) and to shut down the reactor during an AOO is identified as a function performed by safety-related SSCs in the 10 CFR 50.2 definition of safety-related SSCs. (2) The first sentence of GDC 26, states that two independent reactivity control systems of different design principles shall be provided. The third sentence of GDC 26, states that the second reactivity control system shall be capable of reliably controlling the rate of changes resulting from planned, normal power changes (including xenon burnout) to assure specified acceptable fuel design limits are not exceeded. MHTGR-DC 26 (2) is consistent with the current requirements of the second reactivity control system specified in GDC 26. The words "including xenon burnout" have been deleted as this may not be as important for some non-LWR reactor designs. Also, "of different design principles" from the first sentence of GDC 26 has been replaced with "independent and diverse" to clarify the requirement. The reactivity means given by MHTGR-DC 26 (2) is a system important to safety but not necessarily safety-related as it does not mitigate an AOO or accident but is used to control planned, normal reactivity changes such that SARRDLs and the helium pressure boundary design limits are preserved thereby minimizing challenges to the safety-related reactivity control means or protection system.

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Criterion	**MHTGR-DC Title and Content**	**NRC Rationale for Adaptions to GDC**
		The term “independent and diverse” indicates no shared systems or components and a design which is different enough such that no common failure modes exist between the system or means in MHTGR-DC 26 (2) and safety-related systems in MHTGR-DC 26 (1) and (3). (3) Current GDC 27 states that the reactivity control systems shall be designed to have a combined capability 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. Reliably controlling reactivity, as applied to GDC 27 requires that the reactor achieve and maintain a safe, stable condition, including subcriticality, using only safety related equipment with margin for stuck rods (ADAMS Accession No. ML16116A083) (Ref. 29). MHTGR-DC 26 (3) is written to clarify that shut down following a postulated accident using safety-related equipment or means is required. The term “following a postulated accident” refers to the time when plant parameters are relatively stable, no additional means of mitigation are needed and margins to acceptance criteria are constant or increasing. MHTGR-DC 26 allows for a return to power during a postulated accident consistent with the current licensing basis of some existing PWRs if sufficient heat removal capability exists (e.g., PWR main steam line break accident), but MHTGR-DC 26 (1) precludes a return to power during an AOO. (4) The fourth sentence of GDC 26 regarding the capability to reach cold shutdown has been generalized in MHTGR-DC 26 (4) to refer to activities which are performed at conditions below (less limiting than) those normally associated with safe shutdown. SECY-94-084 (Ref. 32) describes staff positions on obtaining a cold shutdown and explains that the requirement to bring the plant

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		to cold shutdown is driven by the need to inspect and repair a plant following an accident. In regards to safety class, the capability to bring the plant to a cold shutdown is not covered by the definition of safety-related SSCs in 10 CFR 50.2, and most operating pressurized-water reactors have not credited safety-related SSCs to satisfy this requirement of GDC 26. Based on the information provided above, the system credited for holding the reactor subcritical under conditions necessary for activities such as refueling, inspection and repair is identified as an important to safety system.
27	*Combined reactivity control systems capability.* Same as ARDC DELETED—Information incorporated into MHTGR-DC 26	
28	*Reactivity limits.* The reactor core, including the reactivity control systems, shall be designed with appropriate limits on the potential amount and rate of reactivity increase to ensure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor helium pressure boundary greater than limited local yielding, nor (2) sufficiently disturb the core, its support structures, or other reactor vessel internals to impair significantly the capability to cool the core.	"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 deleted. Each design will have to determine its postulated reactivity accidents based on the specific design and associated risk evaluation.
29	*Protection against anticipated operational occurrences.* Same as GDC The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their	

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Criterion	**MHTGR-DC Title and Content**	**NRC Rationale for Adaptions to GDC**
	safety functions in the event of anticipated operational occurrences.	

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Criterion	**MHTGR-DC Title and Content**	**NRC Rationale for Adaptions to GDC**
30	*Quality of reactor helium pressure boundary.* Components that are part of the reactor helium 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 leakage. Means shall be provided for detecting ingress of moisture, air, secondary coolant, or other fluids to within the reactor helium pressure boundary.	“Reactor coolant pressure boundary” has been relabeled as “reactor helium pressure boundary” to conform to standard terms used for MHTGRs. The MHTGR-DC 14 addresses the need to consider leakage of contaminants into the helium used to transport heat from the reactor to the heat exchangers for power production, residual heat removal, and process heat. The phrase “reactor helium pressure boundary” encompasses the entire volume containing helium used to cool the reactor, not just the volume within the reactor vessel. For consistency, a specific requirement is appended to MHTGR-DC 30 for a means of detecting ingress of moisture, air, secondary coolant, or other fluids. Although “other fluids” could be interpreted as including water and steam, for emphasis, the word “moisture” is included in the list of contaminants in both MHTGR-DC 14 and MHTGR-DC 30.
31	*Fracture prevention of reactor helium pressure boundary.* The reactor helium pressure boundary shall be designed with sufficient margin to ensure 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, service degradation of material properties, creep, fatigue, stress	“Reactor coolant pressure boundary” has been relabeled as “reactor helium pressure boundary” to conform to standard terms used for MHTGRs. Specific examples are added to the MHTGR DC to account for the high design and operating temperatures, helium coolant, contaminants, and reaction products.

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Criterion	**MHTGR-DC Title and Content**	**NRC Rationale for Adaptions to GDC**
	rupture, 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 and helium composition, including contaminants and reaction products, on material properties, (3) residual, steady-state, and transient stresses, and (4) size of flaws.	
32	*Inspection of reactor helium pressure boundary.* Components that are part of the reactor helium pressure boundary shall be designed to permit (1) periodic inspection and functional testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor vessel.	"Reactor coolant pressure boundary" has been relabeled as "reactor helium pressure boundary" to conform to standard terms used for MHTGRs. The staff modified the LWR GDC by replacing the term "reactor pressure vessel" with "reactor vessel," which the staff believes is a more generically applicable term. A non-leaktight system may be acceptable for some designs provided that (1) the system leakage does not impact safety functions under all conditions, and (2) leakage is consistent with SARRDL. Functional testing is testing that assesses component and system operational readiness such as required in the ASME OM Code as incorporated by reference in 10 CFR 50.55a and in Plant Technical Specifications.
33	*Reactor coolant makeup.* Not applicable to MHTGR.	The MHTGR does not require reactor coolant inventory maintenance for small leaks to meet the SARRDLs, which replaces the concept of the specified acceptable fuel design limits, as discussed in GDC 10. Therefore, ARDC 33 is not applicable to the MHTGR design.

IV. Heat Transport Systems		
Criterion	**MHTGR-DC Title and Content**	**NRC Rationale for Adaptions to GDC**
34	*Passive residual heat removal.* A passive system to remove residual heat shall be provided. For normal operations and anticipated operational occurrences, 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 system radionuclide release design limits and the design conditions of the reactor helium pressure boundary are not exceeded. During postulated accidents, the system safety function shall provide effective cooling. Suitable redundancy in components and features and suitable interconnections, leak detection, and isolation capabilities shall be provided to ensure the system safety function can be accomplished, assuming a single failure.	The word "passive" was added, based on the definition of a MHTGR. In definitions Section 3.1 of the DOE report titled "Guidance for Developing Principal Design Criteria for Advanced (Non-Light-Water) Reactors" (Ref. 17), the MHTGR design has a low power density and hence residual heat is removed by a passive system. "Ultimate heat sink" has been added to explain that, if MHTGR-DC 44 is deemed not applicable to the design, the residual heat removal system is then required to provide the heat removal path to the ultimate heat sink. "Reactor coolant pressure boundary" has been relabeled as "reactor helium pressure boundary" to conform to standard terms used for MHTGRs. The SARRDL replaces the ARDC specified acceptable fuel design limits as described in the rationale to MHTGR-DC 10. The MHTGR-DC 34 incorporates the postulated accident residual heat removal requirements contained in GDC 35. Effective cooling under postulated accident conditions is defined as maintaining fuel temperature limits below design values to help ensure the siting regulatory dose limits criteria at the exclusion area boundary (EAB) and low-population zone (LPZ) are not exceeded and the integrity of the core, the core structural components, and the reactor vessel is maintained under postulated accident conditions, thereby ensuring a geometry required for passive heat removal. The GDC reference to electric power was removed. Refer to the rationale for ARDC 17 on electric power systems.

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Criterion	**MHTGR-DC Title and Content**	**NRC Rationale for Adaptions to GDC**
35	*Emergency core cooling.* Not applicable to MHTGR.	In the MHTGR design maintaining the helium inventory is not necessary to maintain effective cooling. Postulated accident heat removal is accomplished by the residual heat removal system described in MHTGR DC 34.
36	*Inspection of passive residual heat removal system.* The passive residual heat removal system shall be designed to permit appropriate periodic inspection of important components to ensure the integrity and capability of the system.	The word "passive" was added, based on the definition of a MHTGR. In definitions Section 3.1 of DOE report titled "Guidance for Developing Principal Design Criteria for Advanced (Non-Light-Water) Reactors" (Ref. 17), the MHTGR design has a low power density and hence residual heat is removed by a passive system. The GDC 36 system is renamed and revised to provide for inspection of the residual heat removal systems as required for MHTGR-DC 34. The list of examples was deleted, as they apply to LWR designs and each specific design will have different important components associated with residual heat removal.
37	*Testing of passive residual heat removal system*. The passive residual heat removal system shall be designed to permit appropriate periodic functional testing to ensure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the system components, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including associated systems, for AOO or postulated accident decay heat removal to the ultimate heat sink and, if applicable, any system(s) necessary to transition from active normal operation to passive mode.	Criterion 37 has been renamed and revised for testing the passive residual heat removal system required by MHTGR-DC 34. Section 2.3.4 of INL/EXT-10-17997, "Mechanistic Source Terms White Paper," (Ref. 37) notes that the passive reactor cavity cooling system (RCCS) (using either air or water as heat transfer fluid) contributes to the MHTGR safety basis and is subject to component integrity testing. However, Section 6.1 of INL/EXT-11-22708, "Modular HTGR Safety Basis and Approach," (Ref. 38), indicates that RCCS performance does not require "leaktight" conditions. For an RCCS which is an "open system", the normal and expected loss of RCCS coolant through the exhaust structure would not be considered leakage. Abnormal leakage of RCCS coolant to locations other than the exhaust

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Criterion	**MHTGR-DC Title and Content**	**NRC Rationale for Adaptions to GDC**
		structure may be acceptable provided that (1) the RCCS leakage does not impact safety functions under all conditions, and (2) functional containment is not impacted by RCCS leakage. Functional testing is testing that assesses component and system operational readiness such as required in the ASME OM Code as incorporated by reference in 10 CFR 50.55a and in Plant Technical Specifications. The criterion was modified to reflect the passive nature of the MHTGR RCCS to mitigate AOOs or postulated accidents and the need to verify the ability to transition the RCCS from active mode (if present) to passive mode. Some MHTGR 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. Associated systems means testing any auxiliary or secondary systems needed to perform the passive residual heat removal function.
38	*Containment heat removal.* Not applicable to MHTGR.	This criterion is not applicable to the MHTGR. The MHTGR designs do not have a "pressure retaining reactor containment structure" but instead rely on a multibarrier functional containment configuration to control the release of radionuclides. See the MHTGR DC 16 rationale.
39	*Inspection of containment heat removal system.* Not applicable to MHTGR.	This criterion is not applicable to the MHTGR. The MHTGR designs do not have a "pressure retaining reactor containment structure" but instead rely on a multibarrier functional containment configuration to control the release of radionuclides. See the MHTGR-DC 16 rationale.

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Criterion	**MHTGR-DC Title and Content**	**NRC Rationale for Adaptions to GDC**
40	*Testing of containment heat removal system.* Not applicable to MHTGR.	This criterion is not applicable to the MHTGR. The MHTGR designs do not have a "pressure retaining reactor containment structure" but instead rely on a multibarrier functional containment configuration to control the release of radionuclides. See the MHTGR-DC 16 rationale.
41	*Containment atmosphere cleanup.* Not applicable to MHTGR.	This criterion is not applicable to the MHTGR. The MHTGR designs do not have a "pressure retaining reactor containment structure" but instead rely on a multibarrier functional containment configuration to control the release of radionuclides. See the MHTGR-DC 16 rationale.
42	*Inspection of containment atmosphere cleanup systems.* Not applicable to MHTGR.	This criterion is not applicable to the MHTGR. The MHTGR designs do not have a "pressure retaining reactor containment structure" but instead rely on a multibarrier functional containment configuration to control the release of radionuclides. See the MHTGR-DC 16 rationale.
43	*Testing of containment atmosphere cleanup systems.* Not applicable to MHTGR.	This criterion is not applicable to the MHTGR. The MHTGR designs do not have a "pressure retaining reactor containment structure" but instead rely on a multibarrier functional containment configuration to control the release of radionuclides. See the MHTGR-DC 16 rationale.
44	*Structural and equipment cooling.* In addition to the heat rejection capability of the passive residual heat removal system, systems to transfer heat from structures, systems, and components important to safety to an ultimate heat sink shall be provided, as necessary, 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	This renamed MHTGR-DC accounts for advanced reactor system design differences to include cooling requirements for SSCs important to safety, if applicable; this MHTGR-DC does not address the residual heat removal system required under MHTGR-DC 34. The staff inserted "passive" based on the system design for residual heat removal. If a specific MHTGR design can demonstrate that the reactor cavity cooling system (RCCS) provides indefinite core cooling capability, then structural and equipment cooling systems would not be needed.

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Criterion	**MHTGR-DC Title and Content**	**NRC Rationale for Adaptions to GDC**
	be provided to ensure that the system safety function can be accomplished, assuming a single failure.	The GDC reference to electric power was removed. Refer to the rationale for ARDC 17 on electric power systems.
45	*Inspection of structural and equipment cooling systems*. Same as ARDC The 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.	This renamed MHTGR-DC accounts for advanced reactor system design differences to include possible cooling requirements for SSCs important to safety.
46	*Testing of structural and equipment cooling systems.* Same as ARDC The structural and equipment cooling systems shall be designed to permit appropriate periodic functional testing to assure (1) the structural and leaktight integrity of their components, (2) the operability and the performance of the system components, 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 the systems into operation for reactor shutdown and postulated accidents, including operation of associated systems.	This renamed MHTGR-DC accounts for advanced reactor system design differences to include possible cooling requirements for SSCs important to safety. 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. A non-leaktight system may be acceptable for some designs provided that (1) the system leakage does not impact safety functions under all conditions, and (2) defense in depth is not impacted by system leakage. Functional testing is testing that assesses component and system operational readiness such as required in the ASME OM Code as incorporated by reference in 10 CFR 50.55a and in Plant Technical Specifications. "Active" has been deleted in item (2) because appropriate operability and performance tests of system components are required regardless of their active or passive nature. The LOCA reference has been removed to provide for any postulated accident that might affect subject SSCs. The GDC reference to electric power was removed. Refer to the rationale for ARDC 17 regarding electric power systems.

V. Reactor Containment		
Criterion	**MHTGR-DC Title and Content**	**NRC Rationale for Adaptions to GDC**
50	*Containment design basis.* Not applicable to MHTGR.	This criterion is not applicable to the MHTGR. The MHTGR designs do not have a "pressure retaining reactor containment structure" but instead rely on a multibarrier functional containment configuration to control the release of radionuclides. See the MHTGR-DC 16 rationale.
51	*Fracture prevention of containment pressure boundary.* Not applicable to MHTGR.	This criterion is not applicable to the MHTGR. The MHTGR designs do not have a "pressure retaining reactor containment structure" but instead rely on a multibarrier functional containment configuration to control the release of radionuclides. See the MHTGR-DC 16 rationale.
52	*Capability for containment leakage rate testing*. Not applicable to MHTGR.	This criterion is not applicable to the MHTGR. The MHTGR designs do not have a "pressure retaining reactor containment structure" but instead rely on a multibarrier functional containment configuration to control the release of radionuclides. See the MHTGR-DC 16 rationale.
53	*Provisions for containment testing and inspection.* Not applicable to MHTGR.	This criterion is not applicable to the MHTGR. The MHTGR designs do not have a "pressure retaining reactor containment structure" but instead rely on a multibarrier functional containment configuration to control the release of radionuclides. See the MHTGR-DC 16 rationale.
54	*Piping systems penetrating containment.* Not applicable to MHTGR.	This criterion is not applicable to the MHTGR. The MHTGR designs do not have a "pressure retaining reactor containment structure" but instead rely on a multibarrier functional containment configuration to control the release of radionuclides. See the MHTGR-DC 16 rationale.
55	*Reactor coolant boundary penetrating containment.* Not applicable to MHTGR.	This criterion is not applicable to the MHTGR. The MHTGR designs do not have a "pressure retaining reactor containment structure" but instead rely on a multibarrier functional containment

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Criterion	MHTGR-DC Title and Content	NRC Rationale for Adaptions to GDC
		configuration to control the release of radionuclides. See the MHTGR-DC 16 rationale.
56	*Primary Containment isolation.* Not applicable to MHTGR.	This criterion is not applicable to the MHTGR. The MHTGR designs do not have a "pressure retaining reactor containment structure" but instead rely on a multibarrier functional containment configuration to control the release of radionuclides. See the MHTGR-DC 16 rationale.
57	*Closed system isolation valves.* Not applicable to MHTGR.	This criterion is not applicable to the MHTGR. The MHTGR designs do not have a "pressure retaining reactor containment structure" but instead rely on a multibarrier functional containment configuration to control the release of radionuclides. See the MHTGR-DC 16 rationale.

VI. Fuel and Reactivity Control

Criterion	MHTGR-DC Title and Content	NRC Rationale for Adaptions to GDC
60	*Control of releases of radioactive materials to the environment.* Same as GDC 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.	
61	*Fuel storage and handling and radioactivity control.* Same as ARDC	The underlying concept of establishing functional requirements for radioactivity control in fuel storage and fuel handling systems is independent of the design of non-LWR advanced reactors.

VI. Fuel and Reactivity Control		
Criterion	**MHTGR-DC Title and Content**	**NRC Rationale for Adaptions to GDC**
	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 cooling under accident conditions.	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.
62	*Prevention of criticality in fuel storage and handling.* Same as GDC Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.	
63	*Monitoring fuel and waste storage.* Same as GDC 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.	
64	*Monitoring radioactivity releases.* Means shall be provided for monitoring the reactor building atmosphere, effluent discharge paths, and plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.	The underlying concept of monitoring radioactivity releases from the MHTGR particle fuel to the reactor building, effluent discharge paths, and 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

VI. Fuel and Reactivity Control		
Criterion	**MHTGR-DC Title and Content**	**NRC Rationale for Adaptions to GDC**
		protection. Recirculation of loss-of-coolant fluids (i.e., water) does not apply to the MHTGR. The descriptions of the associated atmospheres and spaces that are required to be monitored are revised to reflect the MHTGR's different design configuration and functional containment arrangement.

VII. Additional MHTGR-DC		
Criterion	**MHTGR-DC Title and Content**	**NRC Rationale for Adaptions to GDC**
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.	New MHTGR design-specific GDC are necessary to ensure that the 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 the 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 the release of reactor helium from the building in the event of depressurization accidents.	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 helium pressure boundary. This newly established criterion ensures 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.	This newly established criterion on 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.