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U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Subject: Submittal of Approved Version of GA-EMS Fast Modular Reactor Principal Design Criteria Topical Report

General Atomics - Electromagnetic Systems (GA-EMS) is submitting to the Nuclear Regulatory Commission (NRC) the approved version of the Fast Modular Reactor (FMR) Principal Design Criteria Topical Report. The enclosed topical report does not contain any proprietary information.

The FMR Principal Design Criteria Topical Report was revised per the proposed responses to the six requests for additional information. Those proposed responses were sent to the NRC staff on November 7, 2022 and subsequently reviewed and accepted. The Safety Evaluation of the GA-EMS Principal Design Criteria Topical Report was released by the NRC staff on July 2023.

If you have any questions or need any additional information, please contact John Bolin by email at john.bolin@ga.com or by phone at 858-762-7576.

Sincerely,

John Bolin
Nuclear Technologies and Materials Division
General Atomics – Electromagnetic Systems

Enclosure 1: "Safety Evaluation of Topical Report 30599T00005, Fast Modular Reactor Principal Design Criteria, Revision 2," NRC staff, dated July 6, 2023 (ML22154A555)

Enclosure 2: "Fast Modular Reactor Principal Design Criteria," 30599T00005 Rev. 2-A, August 3, 2023

cc: Samuel.CuadradoDeJesus@nrc.gov
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Enclosure 1:

"Safety Evaluation of Topical Report 30599T00005, Fast Modular Reactor Principal Design Criteria, Revision 2," NRC staff, dated July 6, 2023 (ML22154A555)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

JULY 06, 2023

GENERAL ATOMICS ELECTROMAGNETIC SYSTEMS – SAFETY EVALUATION OF TOPICAL REPORT 30599T00005, FAST MODULAR REACTOR PRINCIPAL DESIGN CRITERIA, REVISION 2 (EPID NO. L-2022-TOP-0033)

SPONSOR AND SUBMITTAL INFORMATION

Sponsor:	General Atomics-Electromagnetic Systems (GA-EMS)
Address:	16530 Via Esprillo San Diego, CA 92127
Project No.:	99902098
Submittal Date:	June 3, 2022
Submittal:	Agencywide Documents Access and Management System (ADAMS) Accession No. ML22154A555

Correspondence Dates and ADAMS Accession Nos:

- U.S. Nuclear Regulatory Commission (NRC) Staff Completeness Determination, dated July 07, 2022, (ML22181B171).
- NRC staff's requests for additional information (RAIs), dated October 05, 2022, (ML22321A310).
- GA-EMS RAI responses, dated November 07, 2022, (ML22311A472).
- GA-EMS Topical Report (TR) 30599T00005, Fast Modular Reactor Principal Design Criteria Revision 2, dated January 5, 2023, (ML23005A290).

1.0 BRIEF DESCRIPTION OF TOPICAL REPORT AND BACKGROUND

By letter dated June 3, 2022 [1], General Atomics – Electromagnetic Systems (GA-EMS) submitted TR 30599T00005, "Fast Modular Reactor Principal Design Criteria," Revision 1 [2], for the NRC staff's review. The TR contains a brief overview of the GA-EMS Fast Modular Reactor (FMR) design, a summary of how the principal design criteria (PDC) were developed, and the PDC selected for the FMR (referred to as the FMR-DC in the TR). GA-EMS requested the NRC staff's review and approval of the FMR PDC TR so it may be referenced by applicants using the FMR design under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 "Domestic Licensing of Production and Utilization Facilities," or 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." Documentation that the PDCs are satisfied will be provided within the license application and is not part of the subject TR.

By email dated July 7, 2022, the NRC staff informed GA-EMS that the TR provided sufficient information for the NRC staff to conduct a detailed technical review [3]. By email dated October 5, 2022, the NRC staff issued RAIs to GA-EMS [4]; GA-EMS responded to the NRC staff's RAIs by letter dated November 7, 2022 [5]. By letter dated January 5, 2023, GA-EMS submitted Revision 2 of the FMR PDC TR [6].

2.0 REGULATORY EVALUATION

The regulations under 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," provide general design criteria (GDC) for water-cooled nuclear power plants similar to those historically licensed by the NRC. Under the provisions of 10 CFR Parts 50 and 52, applicants for a construction permit (CP), operating license (OL), design certification (DC), combined license (COL), standard design approval (SDA), or manufacturing license (ML) must submit PDCs for the proposed facility.

Specifically, the following Commission regulations pertain to the PDCs:

- 10 CFR 50.34(a)(3)(i), which requires, in part, that applications for a CP include PDCs for the facility. An OL would reference a CP, which would include PDCs.
- 10 CFR 52.47(a)(3)(i), which requires, in part, that applications for a DC include PDCs for the facility.
- 10 CFR 52.79(a)(4)(i), which requires, in part, that applications for a COL include PDCs for the facility.
- 10 CFR 52.137(a)(3)(i), which requires, in part, that applications for an SDA include PDCs for the facility.
- 10 CFR 52.157(a), which requires, in part, that applications for an ML include PDCs for the reactor to be manufactured.

The regulations under 10 CFR 50.34(a)(3)(i) state that 10 CFR Part 50, Appendix A, establishes the minimum requirements for the PDCs for water-cooled nuclear power plants similar in design and location to plants for which CPs have previously been issued by the Commission and provides guidance to applicants in establishing PDCs for other types of nuclear power units. Since the GA-EMS FMR is not a water-cooled nuclear power plant, PDCs are required but they do not necessarily align with the minimum requirements in the GDCs in 10 CFR Part 50, Appendix A.

Recognizing that the GDCs in 10 CFR Part 50, Appendix A may not be appropriate for non-light-water reactors (non-LWRs), the NRC issued Regulatory Guide (RG) 1.232, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors," [7] which serves as guidance to develop PDCs for non-LWR designs.

The PDCs are integral to the review of the facility design and should be considered in the development of the facility and the structures, systems, and component (SSC) design bases. PDCs aid in the NRC staff's evaluation of other regulations and allow the NRC staff to have reasonable assurance that the design will conform to the design bases with adequate margins for safety.

3.0 TECHNICAL EVALUATION

3.1 FMR Design Features

As discussed in Section 2, “Design Features of GA-EMS FMR,” of the TR [2] and Section 2.2, “Plant Description” of the GA-EMS regulatory engagement plan [8] the proposed conceptual design for the FMR is a modular high temperature gas-cooled fast spectrum reactor. The reactor is fueled with uranium dioxide pellets loaded into silicon carbide composite clad fuel rods, similar to accident tolerant fuel types proposed for use in operating light-water reactors (LWRs). The fuel rods are arranged in assemblies in a triangular-pitch lattice typical of fast reactor designs, which enables them to be packed more closely than the rectangular-pitch lattice seen in operating LWRs. The primary system coolant is helium. The FMR uses a direct power conversion system with a gas turbine, as shown in Section 2 of the TR [2] and in more detail in Section 2.3, “Power Operation,” of the regulatory engagement plan [8]. Because a direct power conversion system is used, there is no intermediate or secondary loop; the primary system helium coolant is used to drive the turbine to produce electricity. Heat from the power conversion system is transferred to the ultimate heat sink (a dry, forced convection cooling tower) using a water loop.

Rather than a functional containment concept, which is often considered for high temperature gas reactors (HTGRs) (originally considered by the NRC staff in SECY-93-092 [9]), the FMR design includes a leak-tight containment. The reactor and power conversion system are within a structural containment located below grade. Emergency cooling is achieved using a passive system known as the reactor vessel cooling system (RVCS). The RVCS relies on natural circulation of water between structures absorbing heat radiated by the below-grade reactor vessel and an above-grade heat sink to provide continuous heat removal.

In summary, the FMR has different characteristics in common with several different reactor types. The core arrangement, neutron spectrum, and reactor physics are similar to sodium-cooled fast reactors (SFRs), though the FMR fuel is more similar to certain types of accident tolerant fuel proposed for use in LWRs. The reactor coolant system, power conversion system, electric power system, and balance of plant are similar to HTGRs. The use of a leak-tight containment is similar to LWRs and SFRs. The design features of the FMR and their similarity to other types of reactors was used by GA-EMS to inform the development of the FMR PDCs.

3.2 PDC Development Methodology

In Section 3, “FMR PDC Development,” of the TR, GA-EMS stated that the advanced reactor design criteria (ARDCs) in RG 1.232 were used as a starting point for the development of the FMR PDCs. The ARDCs in RG 1.232 were informed by the GDCs and provide guidelines for PDCs for non-LWR designs. The ARDCs are intended to be technology inclusive, and the RG provides technology-specific design criteria for the SFR and the modular high temperature gas reactor (MHTGR). GA-EMS chose to apply both the technology-inclusive ARDCs and technology-specific criteria, as applicable, because the FMR has design elements similar to those used in developing the SFR- and MHTGR-design criteria (SFR-DC and MHTGR-DC, respectively).

To develop the FMR-DCs, GA-EMS refined the ARDCs by adapting and applying them to the FMR design. In refining the design criteria, the underlying safety basis for each GDC was considered. In cases where the ARDC was not fully applicable to the FMR design, GA-EMS considered the SFR-DC and MHTGR-DC. In determining which of the two technology-specific criteria were most relevant for a given PDC, GA-EMS considered which design (SFR or MHTGR) was most similar to the FMR for the SSC(s) covered by the particular PDC in question. The most relevant technology-specific criterion was then used as the basis for the development of the FMR PDC, which in some cases was further adapted to accommodate the details of the FMR design.

The NRC staff considers this overall approach to be acceptable as it uses the NRC staff-approved guidance in RG 1.232 as a basis for developing design-specific criteria applicable to the FMR.

3.3 Evaluation of FMR Principal Design Criteria

Sections 3.1, “Overall Requirements (FMR Design Criteria 1-5),” through 3.6, “Fuel and Radioactivity Control (FMR Design Criteria 60-64),” of the TR provide an overview of the FMR-DC and highlight important decisions made in developing the criteria. Section 4, “FMR Principal Design Criteria,” of the TR provides Table 1 “FMR Principal Design Criteria,” containing all the FMR-DCs and the rationale for adaptations to the GDC for each FMR-DC, similar to tables of PDCs presented in RG 1.232.

3.3.1 Overall Requirements (FMR-DC 1-5)

FMR-DC 1, 2, and 5 provide criteria for quality standards and records, protection against natural phenomena, and sharing of SSCs, respectively. Consistent with the corresponding ARDCs, these PDCs are unchanged from GDCs 1, 2, and 5. The NRC staff finds that these ARDCs 1, 2, and 5 are sufficiently broad to apply to the FMR, and the rationale for the underlying safety basis documented in RG 1.232 remains applicable. As such, the NRC finds FMR-DC 1, 2, and 5 to be acceptable.

FMR-DC 3 provides criteria for fire protection. The language adopted for this PDC is the same as ARDC 3. The NRC staff finds that FMR-DC 3 is acceptable because no further adaptation is needed for ARDC 3 and its underlying safety basis, as documented in RG 1.232, is applicable to the FMR design.

FMR-DC 4 provides criteria for the environmental and dynamic effects design basis. While the design criterion does not exclusively relate to the coolant system design, the differences between ARDC 4, SFR-DC 4, and MHTGR-DC 4 are primarily due to considerations resulting from the coolant system design. For FMR-DC 4, GA-EMS adopted MHTGR-DC 4, which is appropriate considering the FMR utilizes a helium coolant system similar to the MHTGR. Because MHTGR-DC 4 is sufficiently broad to apply to the FMR, and the underlying safety basis documented in RG 1.232 remains applicable, the NRC staff considers FMR-DC 4 to be acceptable.

3.3.2 Protection by Multiple Fission Product Barriers (FMR-DC 10-19)

Section 3.2, "Protection by Multiple Fission Product Barriers," of the TR notes that the FMR fission product barriers include fuel pellets, fuel cladding, reactor vessel, and the containment building and associated systems. The FMR fuel safety design approach seeks to control radionuclides primarily at the source during normal operation and accident conditions. The FMR-DC accordingly use the concept of the specified acceptable fuel design limit (SAFDL), as found in GDC 10.

FMR-DC 10 uses language identical to GDC 10, but replaces the word "coolant" with "heat removal." The FMR uses helium coolant, like the MHTGR. Therefore, this adaptation was chosen to be consistent with MHTGR-DC 10. The rationale for adaptations to the GDC for MHTGR-DC 10 notes the following: "[t]he word 'coolant' has been replaced with 'heat removal,' as helium coolant inventory control for normal operation and AOOs [anticipated operational occurrences] is not necessary to meet the SARRDLs [specified acceptable radiological release design limits], due to the reactor system design." Though GA-EMS does not propose the use of specified acceptable radiological release design limits (SARRDLs) for the FMR-DC, the NRC staff finds that this adaptation of GDC 10 is appropriate because the FMR is a helium-cooled reactor and, consistent with the other FMR-DC, does not need helium coolant inventory control for normal operation and anticipated operational occurrences (AOOs). The NRC staff notes that wholesale adoption of the MHTGR-DC for the FMR design is not appropriate because the FMR-DC use SAFDLs rather than SARRDLs.

FMR-DC 11, which ensures the FMR has desirable inherent feedback characteristics, is the same as ARDC 11. The NRC staff finds FMR-DC 11 acceptable as ARDC 11 is sufficiently broad to apply to the FMR and the rationale for the underlying safety basis documented in RG 1.232 remains applicable.

FMR-DC 12 requires suppression of reactor power oscillations to preserve fuel integrity. The language in FMR-DC 12, as described in the GA-EMS response to RAI FMR-DC 12 [5] and as modified in Revision 2 of the FMR PDC TR [6], is almost identical to MHTGR-DC 12 but replaces SARRDLs with SAFDLs and includes the effects of structures on power oscillations. GA-EMS's rationale for using MHTGR-DC 12 as the basis for FMR-DC 12 is that the helium coolant does not affect core susceptibility to coolant-induced power oscillations, and it is therefore appropriate to remove the word "coolant," as used in GDC 12. The effects of structures associated with the reactor core were added to FMR-DC 12 because items such as reflectors – which may be considered either outside or not part of the core – may affect susceptibility of the core to power oscillations. The addition of "associated structures" reflects the same language used in ARDC 12. Based on the above discussion, the NRC staff finds that FMR-DC 12 appropriately captures potential contributors to reactor power oscillations in a gas-cooled fast spectrum reactor and is, thus, acceptable.

FMR-DC 13 ensures instrumentation will be able to monitor important variables and systems, and controls will be provided to maintain these variables and systems within prescribed operating ranges. In RG 1.232, the primary differences between ARDC 13, SFR-DC 13, and MHTGR-DC 13 relate to the coolant system design and the containment design. FMR-DC 13 adopts the same language as ARDC 13 but replaces the words "reactor coolant boundary" with “reactor helium pressure boundary,” similar to MHTGR-DC 13. The use of “reactor helium pressure boundary” is appropriate given the helium coolant system design for the FMR and does not affect the underlying safety basis documented in RG 1.232, which remains applicable. The NRC staff therefore determined that FMR-DC 13 is acceptable.

FMR-DC 14 and 15 provide design criteria for the reactor helium pressure boundary and adopts the language from MHTGR-DC 14 and 15. The NRC staff finds the use of MHTGR-DC 14 and 15 to be appropriate because the FMR’s helium coolant system design is very similar to that of an MHTGR. Because MHTGR-DC 14 and 15 are appropriate for use with the FMR and the underlying safety basis documented in RG 1.232 remains applicable, the NRC staff determined that FM-DC 14 and 15 are acceptable.

FMR-DC 16 provides a design criterion for the containment. FMR-DC 16 is consistent with SFR-DC 16, which replaces the concept of an “essentially leak-tight” containment from GDC 16 with a concept for a containment “consisting of a low-leakage, pressure retaining structure” with leakage “restricted to 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 basis for SFR-DC 16 is discussed in SECY-93-092 [9], where the Commission agreed that an advanced reactor using a low-leakage, pressure-retaining containment concept should not be required to meet the “essentially leak-tight” statement in GDC 16, provided the containment leakage is less than that needed to meet the acceptable onsite and offsite dose consequence limits. Thus, the NRC staff finds FMR-DC 16 to be consistent with NRC policy for a reactor and containment design of the type used in the FMR and is, therefore, acceptable.

FMR-DC 17 provides design requirements for electric power systems. The NRC staff noted departures from concepts in RG 1.232 for this criterion in the FMR-DC TR. Specifically, MHTGR-DC 17, states, in part, “[i]f 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,” which was deleted from FMR-DC 17. The NRC staff asked GA-EMS in RAI FMR-DC 17(A) [4], to explain the deletion and clarify whether power is required for important to safety functions. In response to the RAI [5], GA-EMS provided additional information and stated that the deleted statement will be added to the FMR-DC consistent with ARDC 17 and MHTGR-DC 17. Consistent with its response to the RAI [5] GA-EMS added the statement to FMR-DC 17 in Revision 2 of its FMR PDC TR [6].

The NRC staff also noted that requirements for an additional power system were not provided, in FMR-DC 17. Specifically, MHTGR-DC 17 states, in part, “[t]he electric power systems shall include an onsite power system and an additional power system. […] An additional power system shall have sufficient independence and testability to perform its safety function.” The NRC staff requested additional information on any additional power systems in RAI FMR-DC 17(B) [4]. In its response to the RAI [5], GA-EMS provided additional information and stated that language regarding the additional power source will be added to the FMR-DC 17 consistent with MHTGR-DC 17. Consistent with its response to the RAI [5], GA-EMS added the statement to FMR-DC 17 in Revision 2 of its FMR PDC TR [6].

Further, the NRC staff noted that the rationale for FMR-DC 17 in Table 1 of the TR, states, in part, that, “The GDC text related to ‘…supplies, including batteries, and the onsite distribution system,’ was deleted to allow increased flexibility in the design of offsite power systems for advanced reactor designs.” The NRC staff requested clarification regarding the deletion in RAI FMR-DC 17(B) [4]. In its response [5], GA-EMS provided additional information and stated that the specific reference to “…supplies, including batteries, and the onsite distribution system,” will be removed from the rationale and replaced with the rationale provided in MHTGR-DC 17. Specifically, GA-EMS added “[t]he 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,” to the rationale for FMR-DC 17. The rationale in MHTGR-DC 17 is applicable to FMR-DC 17 since the FMR is very similar to an MHTGR design for the purposes of the electric power system, as discussed in Section 3.1 of this SE. Consistent with its response to the RAI [5], GA-EMS added the rationale in MHTGR -DC 17 into FMR-DC 17 in Revision 2 of its FMR PDC TR [6]. In conclusion, the NRC staff compared FMR-DC 17 and MHTGR-17 in RG 1.232. The NRC staff verified that FMR-DC 17 and its rationale is the same as that provided in the NRC staff’s guidance for MHTGR-DC 17 in RG 1.232. Because MHTGR-DC 17 is appropriate for use with the FMR and the underlying safety basis documented in RG 1.232 remains applicable, the NRC staff finds FMR-DC 17 to be acceptable.

FMR-DC 18 provides a criterion for inspection and testing of electric power systems. For FMR-DC 18, Table 1 in the TR states that FMR-DC 18 is the same as ARDC 18 and is applicable to the FMR design. The NRC staff compared FMR-DC 18 and ARDC 18 in RG 1.232 and verified that FMR-DC 18 is the same as the ARDC 18 in RG 1.232. Because ARDC 18 is sufficiently broad to apply to the FMR, and the underlying safety basis documented in RG 1.232 remains applicable, the NRC staff finds FMR-DC 18 to be acceptable.

FMR-DC 19 provides control room design criteria. The TR states that FMR-DC 19 is the same as MHTGR-DC 19, but RG 1.232 also states that MHTGR-DC 19 is the same as ARDC 19. ARDC 19 is sufficiently broad to apply to the FMR, and the rationale for the underlying safety basis documented in RG 1.232 remains applicable; therefore, the NRC staff determined that FMR-DC 19 is acceptable.

3.3.3 Protection and Reactivity Control Systems (FMR-DC 20-29)

FMR-DC 20 through 24 provide design criteria for protection system functions, reliability and testability, independence, and failure modes and the separation of protection and control systems. these PDCs are the same as ARDC 20 through 24 (which are the same as GDC 20 through 24). The NRC staff determined that ARDCs 20 through 24 are sufficiently broad to apply to the FMR design without adaptation and their underlying safety basis remains applicable. Therefore, the NRC staff determined that FMR-DC 20 through 24 are acceptable.

FMR-DC 25 provides design criteria for the protection system requirements for control rod malfunctions. FMR-DC 25 and the associated rationale is identical to ARDC 25. Because ARDC 25 is sufficiently broad to apply to the FMR design without adaptation and the underlying safety basis documented in RG 1.232 remains applicable, the NRC staff determined that FMR-DC 25 is acceptable.

FMR-DC 26 provides criteria for the reactivity control system. The language used is identical to ARDC 26, except that the effects of xenon burnout are explicitly mentioned. This deviation from ARDC 26 is acceptable to the NRC staff because it results in a more restrictive criterion, though the effects of xenon are expected to be minimal for fast reactors. This modification to ARDC 26 is minor and does not affect the underlying safety basis documented in RG 1.232; therefore, the NRC staff determined that FMR-DC 26 is acceptable. The NRC staff notes that GA-EMS did not propose an FMR-DC 27, as the criteria of GDC 27 are incorporated into FMR-DC 26, consistent with ARDC 26. This is also acceptable to the NRC staff because the underlying basis achieved by GDC 27 is incorporated into FMR-DC 26.

FMR-DC 28 provides criteria for reactivity limits and is the same as MHTGR-DC 28. This is acceptable to the NRC staff because MHTGR-DC 28 refers to reactivity control limits protecting the reactor helium pressure boundary from postulated reactivity accidents, which is consistent with the FMR design. Because MHTGR-DC 28 is appropriate for use with the FMR and the underlying safety basis documented in RG 1.232 remains applicable, the NRC staff determined that FMR-DC 28 is acceptable.

FMR-DC 29 provides criteria for protection against AOOs. The language used is identical to that in ARDC 29 (which is itself the same as GDC 29), and is acceptable to the NRC staff because it is sufficiently broad such that it is applicable to the FMR without modification and the underlying safety basis documented in RG 1.232 remains applicable.

3.3.4 Fluid Systems (FMR-DC 30-46)

FMR-DC 30 through 32 provide criteria for the reactor helium pressure boundary. The language used for these PDCs is the same as the respective MHTGR-DCs, which refer to the reactor helium pressure boundary because the MHTGR uses helium as its primary coolant. This is acceptable to the NRC staff because helium is also used as the FMR primary coolant. Because MHTGR-DC 30 through 32 apply to the FMR design and the underlying safety basis documented in RG 1.232 remains applicable, the NRC staff determined that FMR-DC 30 through 32 are acceptable.

GA-EMS did not propose a criterion for FMR-DC 33. GDC 33 provides requirements for reactor coolant makeup systems for LWRs. GA-EMS stated in the TR that no similar criterion is applicable for the FMR because reactor coolant inventory makeup is not needed to meet SAFDLs for small leaks. The NRC staff finds that this is consistent with the RG 1.232 rationale for non-applicability of GDC 33 to MHTGRs and is, therefore, acceptable.

FMR-DC 34 provides design criteria for the residual heat removal system. As provided by GA-EMS in its response to RAI FMR-DC 34 [5] and Revision 2 of its FMR PDC TR, the NRC staff notes that the language used is almost identical to MHTGR-DC 34 except it refers to "[s]ystem(s) to remove residual heat" rather than a single "passive system." Similarly, the rationale for adaptations to GDC 34 for FMR-DC 34 is consistent with that provided for MHTGR-DC 34, but states that both active non-safety related systems and passive safety-related systems are available to remove residual heat. Since both active non-safety-related and passive safety-related systems are available in the design, the NRC staff finds that it is reasonable for the FMR-DC to encompass all these systems. Because these adaptations to GDC 34 in MHTGR-DC 34 are appropriate for the FMR design, the NRC staff determined that FMR-DC 34 is acceptable.

GA-EMS did not propose a criterion for FMR-DC 35. GDC 35 provides requirements for emergency core cooling systems (ECCS) for LWRs. GA-EMS stated that ECCS criteria are not applicable to the FMR because it is not necessary to maintain helium coolant inventory to provide emergency core cooling and because postulated accident heat removal is accomplished by the passive residual heat removal system described in FMR-DC 34. This rationale is consistent with the rationale from RG 1.232 regarding emergency core cooling in MHTGRs and therefore is acceptable to the NRC staff.

FMR-DC 36 provides criteria for inspection of the passive residual heat removal system. This PDC is the same as MHTGR-DC 36. This is appropriate because the residual heat removal system described in MHTGR-DC 36 is consistent with the proposed design of the FMR. Because MHTGR-DC 36 is appropriate for use with the FMR, and the underlying safety basis documented in RG 1.232 remains applicable, the NRC staff determined that FMR-DC 36 is acceptable.

FMR-DC 37, as updated in GA-EMS response to RAI FMR-DC 37 and Revision 2 of the FMR PDC TR [5], provides criteria for testing of the residual heat removal systems. FMR-DC 37 is similar to MHTGR-DC 37, with a change to indicate that multiple systems, both passive safety-related systems and active non-safety-related systems, may be relied upon for residual heat removal. This is consistent with FMR-DC 34, regarding the design of the residual heat removal system. As with FMR-DC 34, the NRC staff finds it reasonable to include both active non-safety-related systems and passive safety-related systems in FMR-DC 37 because both types of systems are available in the design to remove residual heat. Because of the consistency with FMR-DC 34 and similarity to MHTGR-DC 37, which is the most relevant criterion from RG 1.232, the NRC staff determined that FMR-DC 37 is acceptable.

FMR-DC 38 through 43 provide criteria related to systems that help maintain containment integrity and enhance performance during and following postulated accidents, including the containment heat removal system and the containment atmosphere cleanup system (and provisions for the inspection and testing of both systems). The language used for these PDCs is the same as the corresponding ARDCs. The NRC staff finds that these PDC are acceptable because the FMR is utilizing a leak-tight containment design, consistent with the bases for these ARDCs.

FMR-DC 44 through 46 provide criteria related to structural and equipment cooling systems, including the inspection and testing of such systems. For all of these PDCs, the language used is the same as the corresponding ARDC. Because ARDCs 44 through 46 are sufficiently broad to apply to the FMR design and the underlying basis documented within RG 1.232 remains applicable, the NRC staff determined that FMR-DC 44 through 46 are acceptable.

3.3.5 Reactor Containment (FMR-DC 50-57)

FMR-DC 50 through 53 provide criteria related to the design of the FMR structural leak-tight containment, including penetrations, and provisions for testing and inspection of portions important for the leak-tight performance of the structure. For these PDCs, the language used is the same as ARDCs 50 through 53. The NRC staff finds that these criteria are acceptable because the FMR is utilizing a leak-tight containment design, consistent with the bases for these ARDC.

FMR-DC 54 describes general criteria for the design of FMR piping systems penetrating containment, including provisions for leak detection, isolation, and testing. FMR-DC 54 is identical to SFR-DC 54, except for the deletion of the word "reactor" from "reactor containment structure" used in SFR-DC 54. GA-EMS described that the word "reactor" was removed from the FMR-DC 54 because the containment is a barrier between fission products and the environment. Because the direct Brayton cycle power conversion system could have fission products in the helium working fluid, this change is appropriate to identify the presence of major SSCs within containment besides the reactor itself. The FMR conceptual design provides containment around the reactor and power conversion system, consistent with the function to contain fission products. Both SFR-DC 54 and FMR-DC 54 differ from ARDC 54 by the replacement of the phrase "having redundancy, reliability, and performance capabilities that reflect the importance to safety of isolating these piping systems" with "that have redundancy, reliability, and 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." As described in RG 1.232 and by GA-EMS, the intent of the change is to accommodate designs capable of demonstrating that containment isolation valves are not necessary for certain piping penetrations that do not provide a credible release path to the atmosphere, such as a closed passive residual heat removal system or intermediate cooling loop. The FMR conceptual design includes the RVCS and intermediate power conversion system heat removal loops that could be designed to achieve the containment function without isolation valves, consistent with the basis for SFR-DC 54. Regardless of the means, FMR-DC 54 specifies that the design capabilities necessary to perform the containment safety function and prevent radioactivity releases from containment will be present. Therefore, the NRC staff finds that FMR-DC 54 is acceptable.

FMR-DC 55 through 57 provide the criteria for the design and configuration of piping penetration isolation capability. FMR-DC 55 and 57 are identical to ARDC 55 and ARDC 57, respectively, except for the replacement of "reactor coolant boundary" with "reactor helium pressure boundary" in FMR-DC 55 (both title and text) and 57 (text only). In its response to RAI FMR-DC 56 [5] GA-EMS stated it would incorporate a modification to FMR-DC 56 that would add the following statement: "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" to make it identical to ARDC 56. Consistent with its response to the RAI, GA-EMS incorporated this statement into FMR-DC 56 in its Revision 2 of the FMR PDC TR [6]. The replacement of "reactor coolant boundary" with "reactor helium pressure boundary" merely describes the FMR coolant. The FMR conceptual design includes a structural containment with piping penetrations that may include isolation valves, which is consistent with the bases for ARDCs 55 through 57. Therefore, the NRC staff finds that FMR-DC 55 through 57 are acceptable.

3.3.6 Fuel and Radioactivity Control (FMR-DC 60-64)

FMR-DC 60 provides criteria for the capability to control releases of radioactive material to the environment during normal reactor operation, including AOOs. FMR-DC 60 is identical to ARDC 60 (which is itself identical to GDC 60). Since the types of radioactive material addressed by ARDC 60 are applicable to the FMR, ARDC 60 is appropriate to use with the FMR and the underlying safety basis documented in RG 1.232 remains applicable. Therefore, the NRC staff determined that FMR-DC 60 is acceptable.

FMR-DC 61 provides criteria for the fuel storage and handling and radioactivity control under normal and postulated accident conditions. FMR-DC 61 is identical to ARDC 61 for fuel storage. The criteria are sufficiently general to be consistent with expected FMR fuel storage and handling operations and are modified from GDC 61 to provide for a variety of methods for fuel cooling. Because ARDC 61 is appropriate for use with the FMR design and the underlying safety basis documented in RG 1.232 remains applicable, the NRC staff determined FMR-DC 61 is acceptable.

FMR-DC 62 provides a criterion for preventing criticality in fuel storage and handling systems. The language used is the same as the ARDC 62 (which is itself the same as GDC 62). Because ARDC 62 is applicable without modification to the FMR design and the underlying safety basis documented in RG 1.232 remains applicable, the NRC staff determined FMR-DC 62 is acceptable.

FMR-DC 63 provides a criterion for monitoring fuel and waste storage. The language used is the same as ARDC 63 (which is itself the same as GDC 63), which is sufficiently broad to apply to the FMR without modification. Because ARDC 63 is applicable and the underlying safety basis documented in RG 1.232 remains applicable, the NRC staff determined FMR-DC 63 is acceptable.

FMR-DC 64 provides a criterion for monitoring releases of radioactivity. The language used is the same as ARDC 64. For this criterion, RG 1.232 provides SFR and MHTGR-specific PDCs, but the ARDC appears to be the best fit for the FMR design since it is consistent with the pressure-retaining containment approach used for the FMR. Because ARDC 64 is sufficiently broad to apply to the FMR and the underlying safety basis documented in RG 1.232 remains applicable, the NRC staff determined that FMR-DC 64 is acceptable.

4.0 CONCLUSION

Based on the above evaluation, the NRC staff concludes that GA-EMS has considered each of the design aspects presented in RG 1.232 and developed a sufficient set of PDCs that are appropriate for establishing requirements for the FMR design. These PDCs establish the necessary design, fabrication, construction, testing, and performance design criteria for safety significant SSCs to provide reasonable assurance that an FMR could be operated without undue risk to the health and safety of the public. The subject TR is therefore suitable for referencing in future licensing applications for the GA-EMS FMR, provided that the plant design is consistent with that discussed in Section 2 of the TR. If the design differs from that discussed in the TR, justification must be provided as to why the PDCs remain applicable.

5.0 REFERENCES

[1] GA-EMS, “Transmittal of GA-EMS Fast Modular Reactor Principal Design Criteria Topical Report,” dated June 03, 2022, (ML22154A557).
[2] GA-EMS, “Fast Modular Reactor Principal Design Criteria,” 30599T00005, Revision 1, dated May 2022, (ML22154A556).
[3] U.S. NRC, “Completeness Determination for General Atomics Fast Modular Reactor Principal Design Criteria Topical Report,” dated July 07, 2022, (ML22181B171).
[4] U.S. NRC, “Transmittal of Requests for Additional Information - General Atomics-Electromagnetic Systems Fast Modular Reactor Principal Design Criteria Topical Report,” dated October 05, 2022, (ML22321A310).
[5] GA-EMS, “Transmittal of Responses to Request for Additional Information on GA-EMS Fast Modular Reactor Principal Design Criteria Topical Report,” dated November 07, 2022, (ML22311A472).
[6] GA-EMS, "Transmittal of Revised GA-EMS Fast Modular Reactor Principal Design Criteria Topical Report," Revision 2, dated January 5, 2023, (ML23005A290).
[7] U.S. NRC, “Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors,” Regulatory Guide 1.232, dated April 2018, (ML17325A611).
[8] General Atomics, “Fast Modular Reactor Pre-Application Regulatory Engagement Plan,” 30599T00003, Revision 2, dated March 2022, (ML22087A510).
[9] U.S. NRC, “Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and Their Relationship to Current Regulatory Requirements,” SECY-93-092, dated April 1993, (ML040210725).

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Date: JULY 06, 2023

Enclosure 2:

"Fast Modular Reactor Principal Design Criteria," 30599T00005 Rev. 2-A, August 3, 2023

NUCLEAR TECHNOLOGIES AND MATERIALS ADVANCED REACTOR CONCEPTS-20

FAST MODULAR REACTOR PRINCIPAL DESIGN CRITERIA

Sponsored by the U.S. Department of Energy
Under Contract # DE-NE0009052

Contractor: General Atomics
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REVISION HISTORY

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ACRONYMS

Acronym	Definition
ANSI/ANS	American National Standards Institute/American Nuclear Society
AOO	Anticipated Operational Occurrence
ARC-20	Advanced Reactor Concepts-20
ARDC	Advanced Reactor Design Criteria
ARDP	Advanced Reactor Demonstration Program
ASME	American Society of Mechanical Engineers
ATF	Accident Tolerant Fuel
DOE	Department of Energy
EAB	Exclusion Area Boundary
ECN	Engineering Change Notice
FHR	Fluoride High-temperature Reactor
FMR	Fast Modular Reactor
FMR-DC	Fast Modular Reactor Design Criteria
GA-EMS	General Atomics Electromagnetic Systems
GDC	General Design Criteria
GCR	Gas-Cooled Reactor
GFR	Gas-cooled Fast Reactor
GT-MHR	Gas Turbine-Modular Helium Reactor
LFR	Lead-cooled Fast Reactor
LOCA	Loss-of-Coolant Accident
LPZ	Low-Population Zone
LWR	Light Water Reactor
MHTGR	Modular High Temperature Gas-cooled Reactor
MHTGR-DC	Modular High Temperature Gas-cooled Reactor Design Criteria
MSR	Molten Salt Reactors
MWe	Megawatt electric
NRC	Nuclear Regulatory Commission
PCS	Power Conversion System
PCU	Power Conversion Unit
PDC	Principal Design Criteria
PWR	Pressurized Water Reactor
RG	Regulatory Guide
RHR	Residual Heat Removal
RIA	Reactivity-Initiated Accident
RVCS	Reactor Vessel Cooling System
SAFDL	Specified Acceptable Fuel Design Limit
SC-MHR	Steam Cycle Modular Helium Reactor
SFR	Sodium-cooled Fast Reactor

Acronym	Definition
SFR-DC	Sodium-cooled Fast Reactor Design Criteria
SSC	Structure, System, and Component
TCG	Turbine-Compressor-Generator

1 INTRODUCTION

General Atomics Electromagnetic Systems (GA-EMS) is developing a 50 MWe helium-cooled Fast Modular Reactor (FMR) [1]. The project has been selected by the U.S. Department of Energy (DOE) for Advanced Reactor Concepts-20 (ARC-20) under Advanced Reactor Demonstration Program (ARDP). The long-term goal is to design, license, and commercialize the FMR plant by the mid-2030s. Early engagement with Nuclear Regulatory Commission (NRC) is an important licensing strategy of the FMR project. As an effort to support the design and a part of the pre-application regulatory engagement plan, GA-EMS is developing Principal Design Criteria (PDC) applicable to the FMR design.

NRC regulations in 10 CFR 50.34(a)(3)(i) require that applicants for a construction permit include the PDC for a facility. Similarly, NRC regulations 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) require that applications for standard design certifications, combined licenses, standard design approvals, and manufacturing licenses include the PDC for a facility. NRC regulations in 10 CFR 50, Appendix A provide General Design Criteria (GDC) that establish the minimum requirements for PDC for Light Water Reactors (LWRs).

While the regulations noted that the GDC were generally applicable to other types of reactor units and were intended to provide guidance in establishing the PDC for such other units, the NRC and DOE established a joint initiative to address the regulatory framework that could apply to non-LWR technologies and specifically, to address the existing GDC, which may not directly apply to non-LWR power plant designs. This effort resulted in the NRC Regulatory Guide (RG) 1.232 [2].

As discussed in RG 1.232, facilities licensed under 10 CFR 50, including both LWRs and non-LWRs, are required to describe the PDC in their preliminary safety analysis report supporting a construction permit application as described in 10 CFR 50.34(a)(3).

Relevant excerpts from RG 1.232 for development of PDC are provided as follows:

- “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, must include the PDC for the facility in their applications.”
- “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 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, Sodium-cooled Fast Reactor Design Criteria (SFR-DC), or Modular High Temperature Gas-cooled Reactor Design Criteria (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.”
- “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.”
- “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.”

GA-EMS requests NRC review and approval of these PDC to be used by applicants of the FMR design for standard design certifications, combined licenses, standard design approvals, and manufacturing licenses under the applicable regulations in 10 CFR 52; or limited work authorizations, construction permits, and operating licenses under 10 CFR 50. The demonstration that the FMR design satisfies these PDC will be provided within the license application documents (e.g., safety analysis reports) required to be submitted by the cited regulations.

2 DESIGN FEATURES OF GA-EMS FMR

The FMR is a Gas-cooled Fast Reactor (GFR), operating at system temperature range of 500 °C to 800 °C. It is a grid-capable power source with a net electric output of 50 MW. The reactor core uses helium coolant and uranium dioxide (UO_2) fuel pellets encapsulated in a silicon carbide (SiC) composite cladding, arranged in a triangular pitch and forming a hexagonal fuel assembly. GA-EMS has pioneered and leads the industry in developing nuclear-grade SiC composite cladding under the Accident Tolerant Fuel (ATF) program [3, 4].

The reactor core is an annular shape surrounded by solid reflector blocks such as zirconium silicide (Zr_3Si_2) and graphite that preserve neutrons and enhance heat transfer. Zr_3Si_2 is a heavy reflector specifically developed for the GFR [5]. GA-EMS has fabricated samples and tested under low temperature and low irradiation condition to confirm the fabrication process and characteristics [6]. This material is favorable in fast reactors to avoid power peaking around the core periphery.

Helium is chemically inert and will not aggravate an accident by contributing to any chemical or nuclear reaction. The use of helium as the coolant in combination with conventional fuel and effective neutron reflector offered enhanced neutronic and thermal efficiencies and several advanced safety characteristics. The major systems and components are underground as illustrated in Figure 1.

The concept of Power Conversion System (PCS) is similar to that of the Power Conversion Unit (PCU) developed for the Gas Turbine-Modular Helium Reactor (GT-MHR) [7]. GA-EMS has also developed a conceptual design of a 65 MWe PCU [8] which will be used as the base model of the 50 MWe PCU of the FMR. The Turbine-Compressor-Generator (TCG) are mounted on an inline vertical configuration. The generator is in a separate, connected vessel at the top of the PCU. A dry-gas shaft seal isolates the helium in the generator from the primary coolant. The generator cavity is maintained at lower pressure to reduce windage losses.

1. Reactor vessel
2. Reactor vessel cooling panel
3. Power conversion unit
4. Generator
5. Recirculator
6. Maintenance cooling system
7. Water tank
8. Spent fuel storage
9. Containment
10. Containment liner
11. Maintenance caps
12. Concrete
13. Ground level

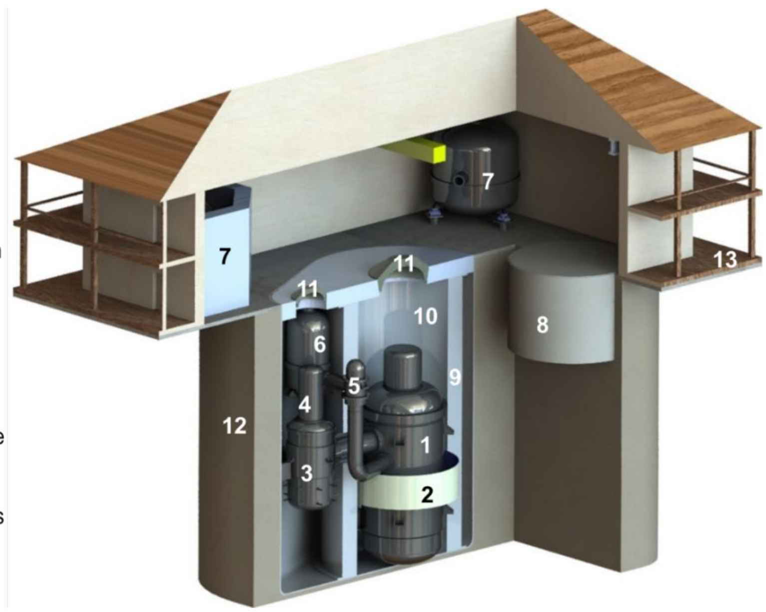

Figure 1. FMR Nuclear Island Components

To achieve the safety objectives for the FMR, the design relies on passive safety features. The FMR is designed to passively remove residual and decay heat from the core regardless of whether helium is present. GA-EMS selected the gravity-driven Reactor Vessel Cooling System (RVCS) because of its reliable passive safety in other Gas-Cooled Reactors (GCRs). Unlike the traditional GCRs [9, 10], packed with solid graphite, the FMR does not rely on conduction-cooldown. For the rodded core like an FMR, the radiation heat transfer, proportional to the temperature to the fourth power, is the dominant heat transfer mechanism from the fuel rods to the surrounding solid structures over conduction or convection. Thus, the passive safety of the core is enhanced by the radiation heat transfer and other design features such as the large thermal margin, low power density, and annular core configuration. The heat from the reactor vessel is transferred by radiation to the cooling panel of the RVCS.

3 FMR PDC DEVELOPMENT

As stated in 10 CFR 50 Part Appendix A, PDC establish the necessary design, fabrication, construction, testing, and performance requirements for Structures, Systems, and Components (SSCs) important to safety. Those SSCs provide reasonable assurance that the nuclear power plant can be operated without undue risk to the health and safety of the public.

RG 1.232 establishes guidance for developing PDC of non-LWR in support of the regulatory requirements. This RG also describes guidance for modifying and supplementing GDC to develop PDC that address non-LWR design concepts in three categories: SFR-DC, MHTGR-DC, and a design-neutral category, Advanced Reactor Design Criteria (ARDC).

The underlying safety objectives of the GDC still apply, as the overall requirements and design criteria for reactivity control systems defined in the GDC are applicable for LWRs and non-LWRs. The ARDC is applicable for the six advanced reactor types: Sodium-cooled Fast Reactors (SFRs), Lead-cooled Fast Reactors (LFRs), GFRs, Modular High Temperature Gas-cooled Reactors (MHTGRs), Fluoride High-temperature Reactors (FHRs), and Molten Salt Reactors (MSRs).

The ARDC from RG 1.232 was used as a starting point for the development of the FMR PDC. The ARDC have been refined by adapting and applying them to a standard FMR design concept. The underlying safety basis for the criterion was considered and rationale for the adaptation was provided to demonstrate how proposed adjustments to the GDC can be translated into qualitative statements of design commitment as a design specific PDC. As described in RG 1.232, in some cases, the ARDC in RG 1.232 adopts the GDC without change. There are also some cases of the ARDC that the NRC rationale for their adaptions to GDC remain valid for the FMR PDC. For those ARDC that did not fully apply to the key design features of the FMR, then the SFR-DC and MHTGR-DC are assessed to determine if either could be directly adopted. If either the SFR-DC or MHTGR-DC are representative of the FMR technology, then the one that is most representative is selected as the FMR PDC.

The development of the FMR PDC is divided into following six sections similar to the GDC in 10 CFR 50.

3.1 Overall Requirements (FMR Design Criteria 1 – 5)

This set of criteria can be generally applied as written for the advanced reactor technologies addressed by RG 1.232. For the FMR, these criteria were left mostly unchanged as compared to the GDC or MHTGR-DC, with minor updates in the rationale for adaptations. FMRs are designed to passively remove residual and decay heat from the core regardless of whether the primary coolant is present. Emphasis on Loss-of-Coolant Accidents (LOCAs) is therefore removed. LOCAs may still require analysis in conjunction with postulated accidents if they are relevant to the design.

Because the FMR design proposes using a direct power cycle, a very high-speed, very high-energy gas-turbine is located inside the reactor helium pressure boundary. The presence of a very high-energy turbine inside the reactor helium pressure boundary creates the potential that a catastrophic dynamic failure of the gas turbine (e.g., at power) could result in the consequential catastrophic failure of the reactor helium pressure boundary caused by the failure of rotating turbine components. This is specifically addressed in FMR-DC 4, i.e., environmental and dynamic effects design bases. The language of prevention, protection, and mitigation of turbine dynamic failure is strengthened to support such PCS design characteristics.

3.2 Protection by Multiple Fission Product Barriers (FMR Design Criteria 10 – 19)

This group of criteria establishes the need for multiple barriers to the release of fission products, consistent with the defense in depth concept for providing independent and redundant layers of defense to compensate for potential human and mechanical failures.

The multiple fission product barriers of the FMR design include the fuel pellets, fuel cladding, reactor vessel, containment building and associated systems. 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 the containment and associated systems.

The FMR fuel safety design approach seeks to control radionuclides primarily at the source, during normal operation and during accident conditions. To meet this objective, the fuel is designed and manufactured to have extremely low levels of initial fabrication defects and to experience very low rates of subsequent incremental failure during normal and postulated accident conditions.

To address the fuel performance, the term chosen to represent the FMR fuel performance limit is Specified Acceptable Fuel Design Limit (SAFDL). During normal operations and Anticipated Operational Occurrences (AOOs), SAFDLs shall not be exceeded. For example, FMR-DC 10 (reactor design) states "the reactor core and associated heat removal, 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." One of the SAFDLs is the fuel cladding temperature limit.

The word "coolant" in GDC 10 has been replaced with "heat removal," as helium coolant inventory control for normal operation and AOOs is not necessary to meet SAFDLs, due to the heat removal mechanism and the reactor system design associated with the PCS. During the normal operation, the core heat is mostly taken away by the convection of the coolant driven by the PCS. However, as the system operating temperature is relatively high, e.g., cladding surface temperature, there always is radiative heat transfer from the fuel rods and heat conduction through the solid structure.

FMR-DC 11 is the same as ARDC 11, which states "the reactor core and associated systems shall be designed so that, in the power operating range, the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity." The word "coolant" has been removed from "reactor core and associated coolant systems" in GDC 11. 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.

FMR-DC 12 is for suppression of reactor power oscillations. It states "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 fuel design limits are not possible or can be reliably and readily detected and suppressed." Helium in the FMR does not affect reactor core susceptibility to coolant-induced power oscillations; therefore, the word "coolant" was deleted for this FMR design criterion.

FMR-DC 14, same as MHTGR-DC 14, addresses the need to consider leakage of contaminants into the helium coolant used to transport heat from the reactor to the heat exchangers for power production, and residual heat removal. The "reactor coolant pressure boundary" in the GDC has been relabeled as "reactor helium pressure boundary" to conform to standard terms used for MHTGR and FMR. 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.

FMR-DC 15 for reactor helium pressure boundary system design is the same as MHTGR-DC 15 because of the similarity of the design in those two reactor concepts.

FMR-DC 16 for containment design is the same as SFR-DC 16 because SFR designs use a low-leakage, pressure-retaining containment concept, similar to the leak-tight containment of the FMR. To provide assurance that the facility can be operated without undue risk to the health and safety of the public, 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.

FMR-DC 17 for electric power systems requires that electric power systems shall be provided to permit functioning of structures, systems, and components important to safety. "Reactor coolant pressure boundary" has been relabeled as "reactor helium pressure boundary" to conform to standard terms used for the FMR.

FMR-DC 18 for inspection and testing of electric power systems is the same as ARDC 18, which is a design-independent companion criterion to ARDC 17.

FMR-DC 19 for control room was expanded to address overall habitability, in addition to retaining the existing requirements associated with radiation protection.

3.3 Protection and Reactivity Control Systems (FMR Design Criteria 20 – 29)

The control of FMR heat generation is accomplished by a large core negative temperature coefficient of reactivity and two independent reactivity control systems. Control rods drop by gravity into the core upon loss of electrical power. An automatic positive control action initiated in response to various accidents, including Reactivity-Initiated Accidents (RIAs), can also cause the rods to drop, or the event itself may cut the power supply. An FMR may not necessarily shut down rapidly (within seconds), but the shutdown should occur in a time frame such that the fission product barrier design limits are not exceeded.

FMR-DC 26 (Reactivity control systems) combines the scope of GDC 26 (Reactivity control system redundancy and capability) and GDC 27 (Combined reactivity control systems capability). The first sentence of current GDC 26 states that two 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. FMR-DC 26 is consistent with the current requirements of the second reactivity control system specified in GDC 26.

FMR-DC 26 implemented changes to the corresponding ARDC language to provide the flexibility to allow for more than two reactivity control systems and to allow any of the available reactivity control systems to provide the capability to keep the reactor subcritical under cold conditions.

3.4 Fluid Systems (FMR Design Criteria 30 – 46)

FMR-DCs 30 to 32 are the same as those design criteria for MHTGR. The “reactor coolant pressure boundary” in the GDC has been relabeled as “reactor helium pressure boundary” to conform to standard terms used for MHTGR and FMR. A specific requirement is appended to FMR-DC 30 for a means of detecting ingress of moisture, air, secondary coolant, or other fluids.

MHTGR-DC 33 for reactor coolant makeup is not applicable to the FMR, as the FMR does not require reactor coolant inventory maintenance for small leaks to meet the SAFDLs.

The LWR “reactor coolant pressure boundary” terminology and other similar system descriptions have been revised to reflect the cooling-related role played by the reactor helium pressure boundary of the FMR. While retention of primary circuit coolant is an important operational function, “core coverage” by the coolant to protect core integrity and inhibit subsequent radionuclide release is not a required safety function of the FMR, because the required safety function of the FMR is to provide structural support for the reactor core and maintain geometry adequate for passive heat removal via radiation and conduction. Therefore, the FMR design criteria dealing with fluid systems had to be modified to emphasize these design attributes.

Like an LWR, the FMR utilizes multiple methods of core heat removal. During normal operations, reactor cooling can be accomplished by utilizing the main loop cooling system. In case that all forced cooling capabilities become unavailable, the overall FMR core design ensures passive residual heat transfer and removal capability that maintains fuel temperatures below design objectives. Passive heat removal performance is achieved regardless of whether the primary reactor circuit is pressurized or depressurized.

FMR-DC 34 is titled as “Passive residual heat removal”. The word “passive” was added based on the FMR design. FMR-DC 34 incorporates the postulated accident residual heat removal requirements contained in GDC 35 (Emergency core cooling). “Ultimate heat sink” has been added in FMR-DC 34 to explain that, if FMR-DC 44 (Structural and equipment cooling) 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. FMR-DC 44 does not address the residual heat removal system required under FMR-DC 34.

The FMR PDC set forth in Criteria 38 – 46, same as ARDCs, presume that a containment structure is used to provide a needed radionuclide retention function and address topics of containment heat removal, atmosphere cleanup, and cooling.

3.5 Reactor Containment (FMR Design Criteria 50 – 57)

These criteria address the design requirements of the reactor containment structure that supports the function of limiting the release of radionuclides to the environment. FMR-DC 50 (Containment design basis) specifically addresses a containment structure in the opening sentence, and FMR-DCs 51–57 support the containment structure's design basis. Therefore, FMR-DCs 51 – 57 use the word "structure" to highlight the containment structure-specific criteria.

The title of FMR-DC 55 is "Reactor helium pressure boundary penetrating containment." The containment is a barrier between the fission products and the environment. The rules for containment penetrations to fulfill containment isolation would apply, without being too prescriptive in the design criteria as to whether it is a primary or secondary or reactor containment. The FMR secondary heat transport system through heat exchangers is a separate closed system that does not allow any direct mixing of secondary fluid with the primary coolant helium. For example, the tubing of the precooler and intercooler piping inside the PCU vessel are a part of the primary coolant boundary. FMR-DC 57 (Closed system isolation valves) addresses closed systems that penetrate containment and was used to address the closed system, such as heat exchange loops, that penetrates containment and is not part of the primary coolant boundary (in its entirety).

3.6 Fuel and Radioactivity Control (FMR Design Criteria 60 – 64)

The overall requirements of GDC described for the control and monitoring of releases of radioactivity to the environment and requirements associated with fuel storage, monitoring, and handling are generally applicable to the FMR. That is, FMR-DC 60, 62, and 63 are the same as those of GDC. However, FMR-DC 61 and 64 were adopted from ARDC 61 and 64, respectively.

ARDC 61 (Fuel storage and handling and radioactivity control) includes some modified wording (relative to the original GDC) to allow for the possibility that some advanced design fuel storage systems may use dry fuel storage. The original GDC wording specifying the need to maintain a "coolant inventory" would not apply for those designs. Therefore, the ARDC language was adopted without further adjustments for the FMR design.

ARDC 64 (Monitoring radioactivity releases) allows for some flexibility in identifying areas where monitoring for radioactivity releases is needed. However, the words "spaces containing components for recirculation of LOCA fluids" was removed in FMR-DC 64 because the FMR design doesn't have components for recirculation of LOCA fluids but may have other similar equipment in spaces where radioactivity should be monitored.

4 FMR PRINCIPAL DESIGN CRITERIA

The PDC of the FMR design are listed in Table 1. For each criterion, the rationale refers to changes made to the GDC. Note that the following:

- To understand the rationale, users of this table need to refer to the appropriate GDC.
- When the criterion of the FMR design is the same as that of the ARDC, SFR-DC, or MHTGR-DC, the rational for adaptions to GDC, that the NRC provided in RG 1.232, is used or partially included.

Table 1. FMR Principal Design Criteria

I. Overall Requirements		
Criterion	**FMR-DC Title and Content**	**Rationale for Adaptions to GDC**
1	*Quality standards and records.* Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.	Same as GDC. This requirement is applicable to all nuclear reactor types.

I. Overall Requirements		
Criterion	**FMR-DC Title and Content**	**Rationale for Adaptions to GDC**
2	*Design bases for protection against natural phenomena.* Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.	Same as GDC. This requirement is applicable to all nuclear reactor types.
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. Non-combustible 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.	Same as ARDC This ARDC as written in RG 1.232 is an adaptation from the GDC. The phrase containing examples where non-combustible and fire-resistant materials must be used has been broadened (from “locations such as the containment and control room” to “locations with structures, systems, or components important to safety”). This criterion contains requirements for fire detection and fighting systems that can be applied for the FMR design.

I. Overall Requirements		
Criterion	**FMR-DC Title and Content**	**Rationale for Adaptions to GDC**
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 piping rupture is extremely low under conditions consistent with the design basis for the piping.	Same as MHTGR-DC 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 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. Because the FMR design proposes using a direct power cycle, a very high-speed, very high-energy gas turbine is located inside the reactor helium pressure boundary. The presence of a very high-energy turbine inside the reactor helium pressure boundary creates the potential that a catastrophic dynamic failure of the turbine (e.g., at power) could result in the consequential catastrophic failure of the reactor helium pressure boundary caused by the failure of rotating turbine components. The word of “missiles” is changed to “missiles originating both inside and outside the reactor helium pressure boundary”.
5	*Sharing of structures, systems, and components.* Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.	Same as GDC This GDC is applicable to all reactor technologies. Sharing among nuclear power units is generally not allowed for structures, systems, and components important to safety.

II. Multiple Barriers		
Criterion	**FMR-DC Title and Content**	**Rationale for Adaptions to GDC**
10	*Reactor design.* The reactor core and associated heat removal, 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.	Design features within the FMR reactor system must ensure that the Specified Acceptable Fuel Design Limits (SAFDLs) are not exceeded during normal operations and Anticipated Operational Occurrences (AOOs). The word "coolant" is replaced with "heat removal," as helium coolant inventory control for normal operation and AOOs is not necessary to meet SAFDLs, due to the reactor system design. The FMR design ensures a passive residual heat removal capability, which is not dependent on forced helium circulation.
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.	Same as ARDC The wording has been changed in ARDC 11 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. ARDC 11 is applicable to the FMR design.
12	*Suppression of reactor power oscillations.* The reactor core, associated structures, and associated 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. Helium in the FMR does not affect reactor core susceptibility to coolant-induced power oscillations; therefore, the word "coolant" was deleted.
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 the containment and associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.	"Reactor coolant pressure boundary" in the GDC has been relabeled as "reactor helium pressure boundary" in FMR-DC to conform to standard terms used for MHTGRs and the FMR.

II. Multiple Barriers		
Criterion	**FMR-DC Title and Content**	**Rationale for Adaptions to GDC**
14	*Primary 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.	Same as MHTGR-DC "Reactor coolant pressure boundary" in the GDC has been relabeled as "reactor helium pressure boundary" to conform to standard terms used for MHTGRs and the FMR. The FMR-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, and residual heat removal. 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 FMR-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 FMR-DC 14 and FMR-DC 30.
15	*Reactor helium pressure boundary system 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.	Same as MHTGR-DC "Reactor coolant system" in the GDC has been relabeled as "reactor helium pressure boundary" to conform to standard terms used for MHTGRs and the FMR. The reactor helium pressure boundary is not an individual system, but rather consists of parts of several systems.

II. Multiple Barriers		
Criterion	**FMR-DC Title and Content**	**Rationale for Adaptions to GDC**
16	*Containment design.* A reactor containment consisting of a high-strength, 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.	Same as SFR-DC 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. This concept is applicable to the FMR containment design.

II. Multiple Barriers		
Criterion	**FMR-DC Title and Content**	**Rationale for Adaptions to GDC**
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) specified acceptable fuel 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 their 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 and 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 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 that 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 of paragraph one and the design criteria of paragraph two. 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 and the FMR.

II. Multiple Barriers		
Criterion	**FMR-DC Title and Content**	**Rationale for Adaptions to GDC**
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.	Same as ARDC ARDC 18 is a design-independent companion criterion to ARDC 17. The text related to the nuclear power unit, offsite power system, and onsite power system was deleted to be consistent with ARDC 17. It is applicable for the FMR design.
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 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.	Same as MHTGR-DC The criterion was updated to remove specific emphasis on LOCAs, which may be not appropriate for advanced designs such as the FMR. 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. A control room habitability requirement beyond that associated with radiation protection has been added to address the concern that non-radionuclide accidents may also affect control room access and occupancy.

III. Reactivity Control		
Criterion	**FMR-DC Title and Content**	**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 assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.	Same as GDC
21	*Protection system reliability and testability.* The protection system shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.	Same as GDC
22	*Protection system independence.* The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.	Same as GDC

III. Reactivity Control		
Criterion	**FMR-DC Title and Content**	**Rationale for Adaptions to GDC**
23	*Protection system failure modes.* The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.	Same as GDC
24	*Separation of protection and control systems.* The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.	Same as GDC
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.	Same as ARDC In ARDC, text has been added to GDC 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 FMR-DC technology neutral.

III. Reactivity Control		
Criterion	**FMR-DC Title and Content**	**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 (including xenon burnout) 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.	FMR-DC 26 combines the scope of GDC 26 and GDC 27. (1) Current GDC 26, second sentence, 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 NRC staff recognizes that specifying control rods may not be suitable for advanced reactors. Additionally, reliably controlling reactivity, as required by GDC 26, has been interpreted as ensuring the control rods are capable of rapidly (i.e., within a few seconds) shutting down the reactor. The NRC 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). Additionally, “specified acceptable fuel design limits” is replaced with “design limits for fission product barriers” to be consistent with the AOO acceptance criteria associated with FMR-DC 10 (specified acceptable fuel design limits) and FMR-DC 15 (reactor helium pressure boundary). A non-LWR may not necessarily shut down rapidly (within seconds) but the shutdown should occur in a time frame such that the fission product barrier design limits are not exceeded. In regard to safety class, the capability to shut down the reactor is identified as a function performed by safety-related SSCs in the 10 CFR 50.2 definition of safety-related SSCs.

III. Reactivity Control		
Criterion	**FMR-DC Title and Content**	**Rationale for Adaptions to GDC**
26 (cont.)		(2) Current GDC 26, first sentence, states that two 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. FMR-DC 26 (2) is consistent with the current requirements of the second reactivity control system specified in GDC 26. 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 FMR-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 the specified acceptable fuel design limits and the helium pressure boundary design limits 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 FMR-DC 26 (2) and safety-related systems in FMR-DC 26 (1) and (3).

III. Reactivity Control		
Criterion	**FMR-DC Title and Content**	**Rationale for Adaptions to GDC**
26 (cont.)		(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 required by GDC 27, requires that the reactor achieve and maintain safe, stable conditions, including subcriticality, using only safety related equipment with margin for stuck rods. FMR-DC 26 (3) refers to the safety-related means (systems and/or mechanisms) to achieve and maintain safe shutdown. 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. FMR-DC 26 (3) 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 FMR-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 FMR-DC 26 (4) to refer to activities which are performed at conditions below (less limiting than) those normally associated with safe shutdown.
27	DELETED and incorporated into FMR-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.	Same as MHTGR-DC “Reactor coolant pressure boundary” has been relabeled as “reactor helium pressure boundary” to conform to standard terms used for MHTGRs and the FMR. 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.

III. Reactivity Control		
Criterion	**FMR-DC Title and Content**	**Rationale for Adaptions to GDC**
29	*Protection against anticipated operational occurrences.* The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.	Same as GDC

IV. Fluid Systems		
Criterion	**Criterion**	**Criterion**
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.	Same as MHTGR-DC “Reactor coolant pressure boundary” has been relabeled as “reactor helium pressure boundary” to conform to standard terms used for MHTGRs and the FMR. The FMR-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 FMR-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 FMR-DC 14 and FMR-DC 30.

IV. Fluid Systems		
Criterion	**Criterion**	**Criterion**
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 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 chemistry on material properties, (3) residual, steady-state, and transient stresses, and (4) size of flaws.	Same as MHTGR-DC “Reactor coolant pressure boundary” has been relabeled as “reactor helium pressure boundary” to conform to standard terms used for MHTGRs and the FMR. Specific examples are added to the FMR-DC to account for the high design and operating temperatures and unique potential coolants.
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 leak-tight integrity, and (2) an appropriate material surveillance program for the reactor vessel.	Same as MHTGR-DC “Reactor coolant pressure boundary” has been relabeled as “reactor helium pressure boundary” to conform to standard terms used for MHTGRs and the FMR. The NRC 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-leak-tight 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 specified acceptable fuel design limits.
33	*Reactor coolant makeup.* Not applicable to FMR	Same as MHTGR-DC The FMR does not require reactor coolant inventory maintenance for small leaks to meet the specified acceptable fuel design limits. Therefore, GDC 33 is not applicable to the FMR design.

IV. Fluid Systems		
Criterion	**Criterion**	**Criterion**
34	*Residual heat removal.* System(s) 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 fuel 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 core 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.	For normal operations, anticipated operational occurrences, and postulated accidents, active non-safety-related systems and passive safety-related systems are available to remove residual heat. "Ultimate heat sink" has been added because the residual heat removal systems are required to provide the heat removal path to the ultimate heat sink rather than rely on an additional intermediary system. "Reactor coolant pressure boundary" has been relabeled as "reactor helium pressure boundary" to conform to standard terms used for MHTGRs and the FMR. The FMR-DC 34 incorporates the postulated accident residual heat removal requirements contained in GDC 35. Effective core 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 FMR-DC 17 concerning those systems that require electric power.
35	*Emergency core cooling.* Not applicable to FMR	In the FMR design, maintaining the helium coolant inventory is not necessary to maintain effective core cooling. Postulated accident heat removal is accomplished by the passive residual heat removal system described in FMR-DC 34.

IV. Fluid Systems		
Criterion	**Criterion**	**Criterion**
36	*Inspection of passive residual heat removal system.* The passive residual heat removal shall be designed to permit appropriate periodic inspection of important components to ensure the integrity and capability of the system.	Same as MHTGR-DC The GDC 36 system was renamed and revised to provide for inspection of the passive heat removal systems as required for FMR-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 residual heat removal system.* The residual heat removal system(s) shall be designed to permit appropriate periodic functional testing to ensure (1) the structural and leak-tight integrity of its components, (2) the operability and performance of the system components, and (3) the operability of the system as a whole and, under conditions as close to 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 revised for testing the residual heat removal system(s) required by FMR-DC 34. For normal operations, anticipated operational occurrences, and postulated accidents, active non-safety-related systems and passive safety-related systems are available to remove residual heat. Abnormal leakage of RHR coolant may be acceptable provided that (1) the RHR leakage does not impact safety functions under all conditions, and (2) containment is not impacted by RHR leakage. Functional testing is testing that assesses component and system operational readiness. The criterion was modified to reflect the nature of the FMR residual heat removal systems to mitigate AOOs or postulated accidents and the need to verify the ability to transition from active mode (if present) to passive mode. Associated systems means testing any auxiliary or secondary systems needed to perform the passive heat removal function.
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.	Same as ARDC

IV. Fluid Systems		
Criterion	**Criterion**	**Criterion**
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.	Same as ARDC
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 leak-tight integrity of its components, (2) the operability and performance of the system components, and (3) the operability of the system as a whole, and under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including the operation of associated systems.	Same as ARDC In ARDC, 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. A non-leak-tight system may be acceptable for the FMR design 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, 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 FMR-DC 17 concerning those systems that require electric power.

IV. Fluid Systems		
Criterion	**Criterion**	**Criterion**
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 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.	Same as ARDC 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. The GDC reference to electric power was removed. Refer to FMR-DC 17 concerning those systems that require electric power.
42	*Inspection of containment atmosphere cleanup systems.* The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems.	Same as GDC

IV. Fluid Systems		
Criterion	**Criterion**	**Criterion**
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 leak-tight 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.	Same as ARDC “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 FMR-DC 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-leak-tight 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 GDC reference to electric power was removed. Refer to FMR-DC 17 concerning those systems that require electric power
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 be provided to ensure that the system safety function can be accomplished, assuming a single failure.	Same as MHTGR-DC This renamed DC accounts for advanced reactor design system differences to include cooling requirements for SSCs, if applicable; this DC does not address the residual heat removal system required under FMR-DC 34. The GDC reference to electric power was removed. Refer to FMR-DC 17 concerning those systems that require electric power.

IV. Fluid Systems		
Criterion	**Criterion**	**Criterion**
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.	Same as ARDC This renamed 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.* The structural and equipment cooling systems shall be designed to permit appropriate periodic functional testing to ensure (1) the structural and leak-tight 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.	Same as ARDC This renamed 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-leak-tight 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) 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 FMR-DC 17 concerning those systems that require electric power.

V. Reactor Containment		
Criterion	**FMR-DC Title and Content**	**Rationale for Adaptions 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.	Same as ARDC FMR-DC 50 specifically addresses a containment structure in the opening sentence and FMR-DC 51–57 support the containment structure's design basis. Therefore, FMR-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.
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.	Same as ARDC FMR-DC 51–57 support FMR-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 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 NRC staff believes 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 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 at containment design pressure.	Same as ARDC FMR-DC 51–57 support FMR-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 DC to clearly convey the understanding that this criterion applies to designs employing containment structures.

V. Reactor Containment		
Criterion	**FMR-DC Title and Content**	**Rationale for Adaptions to GDC**
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.	Same as ARDC FMR-DC 51–57 support FMR-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 DC to clearly convey the understanding that this criterion applies to designs employing containment structures.
54	*Piping systems penetrating containment.* Piping systems penetrating the containment structure shall be provided with leak detection, isolation, and containment capabilities that have redundancy, reliability, and 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.	FMR-DC 51–57 support FMR-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 DC 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. Not all penetrations will provide a release path to the atmosphere. Piping that may be of interest in the case of an FMR design is for the intermediate heat transport system and the passive residual heat removal system. A designer may be able to satisfactorily demonstrate that containment isolation valves are not required for an FMR design. This rewording for the FMR-DC provides a designer the opportunity to present the safety case without containment isolation valves and the associated need for testing.

V. Reactor Containment		
Criterion	**FMR-DC Title and Content**	**Rationale for Adaptions to GDC**
54 (cont.)		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 SFR designs. The same modification is applicable to the FMR design. 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. 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	**FMR-DC Title and Content**	**Rationale for Adaptions to GDC**
55	*Reactor helium pressure boundary penetrating containment.* Each line that is part of the reactor helium pressure 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 in-service 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.	FMR-DC 51–57 support FMR-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 DC to clearly convey the understanding that this criterion applies to designs employing containment structures. In some cases, the word “the” was also added to make the phrase grammatically correct. The word “reactor” was removed because the containment is a barrier between the fission products and the environment. The rules for containment penetrations to fulfill containment isolation would apply. How this is accomplished should be left to the designer of the reactor, 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. The title of FMR-DC 55 is the “Reactor helium pressure boundary penetrating containment.” The FMR intermediate heat transport system is a separate closed system that does not allow any direct mixing of intermediate fluid with the primary coolant helium. The tubing of the precooler and associated intermediate loop piping inside the vessel are a part of the primary coolant boundary. FMR-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 heat transfer 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.

V. Reactor Containment		
Criterion	**FMR-DC Title and Content**	**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.	Same as ARDC FMR-DC 51–57 support FMR-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 DC to clearly convey the understanding that this criterion 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.

V. Reactor Containment		
Criterion	**FMR-DC Title and Content**	**Rationale for Adaptions to GDC**
57	*Closed system isolation valves.* Each line that penetrates the containment structure and is neither part of the reactor helium pressure 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 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.	FMR-DC 51–57 support FMR-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 DC 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. 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.

VI. Fuel and Radioactivity Control		
Criterion	**FMR-DC Title and Content**	**Rationale for Adaptions to GDC**
60	*Control of releases of radioactive materials to the environment.* The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.	Same as GDC

VI. Fuel and Radioactivity Control		
Criterion	**FMR-DC Title and Content**	**Rationale for Adaptions to GDC**
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.	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. 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.* Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.	Same as GDC
63	*Monitoring fuel and waste storage.* Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.	Same as GDC
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.	Same as ARDC 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.

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