University of Illinois Urbana-Champaign High Temperature Gas-cooled Research Reactor: Micro Modular Reactor (MMR™) Principal Design Criteria

TOPICAL REPORT

Release 02

Issued by

ULTRA SAFE NUCLEAR

to

The University of Illinois Urbana-Champaign

for

USNRC Project No. 99902094

April 10, 2024

Approved: July 25, 2024

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

July 25, 2024

Dr. Caleb S. Brooks, Associate Professor
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SUBJECT: UNIVERSITY OF ILLINOIS AT URBANA-CHAMPAIGN – SAFETY EVALUATION FOR TOPICAL REPORT RELATED TO PRINCIPAL DESIGN CRITERIA (EPID: L-2023-NFN-0013)

Dear Dr. Brooks:

By letter dated November 15, 2023 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML23319A407), the University of Illinois at Urbana-Champaign (UIUC) submitted the topical report (TR) “University of Illinois Urbana-Champaign High Temperature Gas-cooled Research Reactor: Micro Modular Reactor (MMR™) Principal Design Criteria,” Release 1 (i.e., Revision 0), for U.S. Nuclear Regulatory Commission (NRC) staff review. On April 5, 2024, the NRC staff held a public meeting with UIUC to discuss NRC staff questions related to the UIUC TR (see public meeting summary dated May 8, 2024 (ML24128A254)). In response to the public meeting, by letter dated April 10, 2024 (ML24101A413), UIUC submitted Release 2 (i.e., Revision 1) of the TR.

The NRC staff’s safety evaluation (SE) for the TR, Release 2, is enclosed. The enclosed SE will be made publicly available.

The NRC staff requests that UIUC publish an accepted version of the TR within 3 months of receipt of this letter. The accepted version should incorporate this letter and the enclosed SE after the title page. The accepted version should include a "-A" (designating “accepted”) following the TR identifier.

If you have any questions, please contact Edward Helvenston at (301) 415-4067, or by email at Edward.Helvenston@nrc.gov.

Sincerely,

Holly D. Cruz Signed by Cruz, Holly on 07/25/24

Holly D. Cruz, Acting Chief
Non-Power Production and Utilization Facility Licensing Branch
Division of Advanced Reactors and Non-Power Production and Utilization Facilities
Office of Nuclear Reactor Regulation

Project No.: 99902094

Enclosure:
As stated

cc: via GovDelivery Subscribers

SUBJECT: UNIVERSITY OF ILLINOIS AT URBANA-CHAMPAIGN – SAFETY EVALUATION FOR TOPICAL REPORT RELATED TO PRINCIPAL DESIGN CRITERIA (EPID: L-2023-NFN-0013) DATED: JULY 25, 2024

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ADAMS Accession No.: ML24155A168 **NRR-106**

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UNIVERSITY OF ILLINOIS AT URBANA-CHAMPAIGN – SAFETY EVALUATION OF RELEASE 2 OF TOPICAL REPORT "UNIVERSITY OF ILLINOIS URBANA-CHAMPAIGN HIGH-TEMPERATURE GAS-COOLED RESEARCH REACTOR: MICRO MODULAR REACTOR (MMR™) PRINCIPAL DESIGN CRITERIA" (EPID: L-2023-NFN-0013)

APPLICANT INFORMATION

Applicant: University of Illinois at Urbana-Champaign

Applicant Address: 104 South Wright St.
Urbana, Illinois 61801

Docket /Project No(s).: 99902094

APPLICATION INFORMATION

Submittal Date: November 15, 2023

Submittal Agencywide Documents Access and Management System (ADAMS) Accession No.: ML23319A407 (Reference 1)

Supplement ADAMS Accession No.: ML24101A413 (Reference 2)

Brief Description of the Topical Report: The topical report (TR) "University of Illinois Urbana-Champaign High Temperature Gas-cooled Research Reactor: Micro Modular Reactor (MMR™) Principal Design Criteria," Release 2 (i.e., Revision 1) (Reference 2), describes the principal design criteria (PDC) for the University of Illinois at Urbana-Champaign (UIUC) Micro Modular Reactor (MMR) research reactor. TR sections 1.0, 1.2, and 5.0 state that the UIUC MMR, designed by Ultra Safe Nuclear Corporation (USNC), would use these PDCs to satisfy Title 10 of the *Code of Federal Regulations* (10 CFR) 50.34(a)(3)(i), which requires that applications for a construction permit (CP) include the PDCs for a proposed facility in the preliminary safety analysis report (PSAR) submitted with a CP application. TR section 1.0 states that details of how the MMR design satisfies the PDCs will be described in future licensing submittals. A preliminary high-level description of the planned UIUC MMR design is given in the TR section 2.0 to provide context for the development and UIUC's request for approval of appropriate PDCs. Specifically, the design consists of a modular high temperature gas-cooled reactor (MHTGR) fueled with a tri-structural isotropic (TRISO) based fuel form referred to as Fully Ceramic Micro-encapsulated (FCM) fuel. TR section 1.0 states that the PDCs in the TR are based on the key design features of the USNC MMR technology as well as the MHTGR design criteria provided in U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.232, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors," Revision 0 (Reference 3).

Enclosure

For additional details on the submittal, please refer to the documents located at the ADAMS Accession numbers identified above.

EVALUATION CRITERIA

As discussed in UIUC's Regulatory Engagement Plan (REP) submitted to the NRC by letter dated June 26, 2023 (Reference 4), UIUC plans to apply to license its MMR under the provisions of 10 CFR 50.21(c), as a class 104(c) research reactor. Specifically, UIUC plans to submit CP and operating license applications to the NRC in the future to request authorization for construction and operation, respectively, of the UIUC MMR. The regulations in 10 CFR 50.34 require that applicants for construction permits and operating licenses include a PSAR and a final safety analysis report, respectively, and describe the information that shall be included in these reports. The regulation in 10 CFR 50.34(a)(3)(i) requires, in relevant part with respect to the PDC TR, that CP applicants establish PDCs. The regulation in 10 CFR 50.34(a)(3)(i) also states that 10 CFR Part 50, Appendix A, "establishes minimum requirements for the [PDCs] for water-cooled nuclear power plants … and provides guidance to applicants for [CPs] in establishing [PDCs] for other types of nuclear power units." However, according to the PDC TR, the UIUC MMR would be a research reactor, and would also not be a water-cooled reactor. Therefore, the NRC staff finds that the specific PDC requirements (i.e., general design criteria) in Appendix A to 10 CFR Part 50 would not apply to the UIUC MMR. As such, UIUC is not required to meet the GDC in Appendix A to Part 50. UIUC would still be required, however, to submit PDCs along with a Part 50 CP application.

NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors" (Reference 5), Part 1, "Format and Content," and Part 2, "Standard Review Plan and Acceptance Criteria," provide guidance for preparing and reviewing, respectively, applications for non-power reactors, including research reactors. The most relevant guidance with respect to this TR is contained in chapter 3, "Design of Structures, Systems, and Components," of Parts 1 and 2. In this chapter, NUREG-1537 describes, in part, the identification and development of principal architectural and engineering design criteria for the structures, systems, and components (SSCs) that ensure reactor facility safety and protection of the public.

Guidance regarding non-light-water-reactor (non-LWR) design criteria is contained in RG 1.232, Revision 0 (Reference 3). RG 1.232 outlines example PDCs formulated to be generally applicable to non-LWR designs. Appendix C of RG 1.232 contains a set of PDCs specifically tailored to MHTGR designs, such as the UIUC MMR. In the PDC TR, UIUC references the guidance in RG 1.232 and largely adopts the PDCs in appendix C of RG 1.232, with a limited number of changes to reflect certain aspects of the MMR design. While the example PDCs in RG 1.232 were developed with a focus on nuclear power reactors, the NRC staff considers the guidance adequate for use in developing research reactor PDCs and considers RG 1.232 to be appropriately consistent with the guidance in NUREG-1537. RG 1.232 is therefore appropriate to use to demonstrate that PDCs meet the requirements of 10 CFR 50.34(a)(3)(i) as applicable to research reactors.

SECY-18-0096, "Functional Containment Performance Criteria for Non-Light-Water-Reactors" (Reference 6), which the Commission approved in SRM-SECY-18-0096 (Reference 7), details criteria that describe how advanced reactor designs may justify the use of a functional containment consisting of multiple barriers to limit the release of radioactive materials, in lieu of a traditional LWR-style containment design. As discussed in TR sections 2.0 and 4.0, the UIUC

MMR design relies on this functional containment approach and the associated PDCs reflect this concept.

The regulations in 10 CFR Part 20 establish standards for protection against ionizing radiation resulting from NRC-licensed activities. Based on UIUC's proposed PDCs, the NRC staff expects that in future licensing submittals the UIUC MMR design, including analyses of SSCs identified to mitigate postulated accident conditions, will be assessed against criteria that will be established by UIUC to ensure that applicable regulatory limits regarding radiological consequences (e.g., dose limits in 10 CFR Part 20) will be met.

TECHNICAL EVALUATION

Scope of Review

TR section 2.0 describes the proposed design of the UIUC MMR facility and TR section 3.0 describes the methodology used in developing the PDC for the described design. The NRC staff notes that these sections provide meaningful context for the PDCs proposed in TR section 4.0 and further outlined in appendix A to the TR, which provides a comparison and highlights differences between the example MHTGR PDCs (i.e., MHTGR-DCs) in appendix C of RG 1.232 versus the proposed UIUC MMR PDCs. However, the NRC staff's review and conclusions in this safety evaluation (SE) pertain specifically to the proposed PDCs provided in TR section 4.0 and listed in appendix A of the TR. The NRC staff makes no determinations regarding the acceptability of the preliminary design described in TR section 2.0, the PDC development information detailed in TR section 3.0, or information in other sections of the TR other than the PDCs themselves. The NRC staff also does not make any determinations regarding the proposed design's adherence to the PDCs. This review scope aligns with the action that UIUC requested in PDC TR section 1.4.

UIUC MMR Design Features

This SE section generally summarizes the preliminary design of the planned UIUC MMR for context. This section does not evaluate or make any determination regarding the acceptability of the design itself.

As discussed in TR section 2.0 and UIUC's REP submitted June 26, 2023 (Reference 4), the proposed UIUC MMR would be a 10 megawatt-thermal, uranium fueled, MHTGR that is normally cooled by helium gas that transfers the generated power to a secondary molten salt loop. The reactor core consists of uranium oxy-carbide fueled TRISO particles embedded in a silicon carbide matrix, forming FCM pellets. The FCM pellets will be stacked within solid graphite blocks to make up the fueled reactor core sections. Helium gas will be pumped through the core to transfer the heat generated to a secondary heat transfer system that consists of an intermediate heat exchanger with a molten salt loop.

TR section 2.0 states that the reactor and core SSCs will be designed such that safety-related core cooling can be adequately achieved through passive means that do not require helium coolant, electrical power, or operator action. The NRC staff notes that, based on the proposed PDCs, the design would ensure that specified acceptable radionuclide release design limits (SARRDLs) will be met through reliance on inherent reactivity feedback, functional containment provided by the FCM fuel form, and passive transfer of heat from the reactor core through the vessel, surrounding below-grade building (citadel), and other associated passive SSCs to the surrounding environment.

UIUC MMR PDCs

PDC TR sections 4.1 through 4.5 list the proposed PDCs for the UIUC MMR. Further, appendix A of the PDC TR provides a direct comparison between the proposed UIUC MMR PDCs and the example MHTGR PDCs (i.e., MHTGR-DCs) provided in appendix C of RG 1.232. Appendix A of the TR also discusses any departures from the RG 1.232 PDC verbiage that UIUC proposed in its MMR PDCs.

UIUC MMR PDCs with Minimal or No Departure from RG 1.232

Many of the PDCs UIUC proposed for the UIUC MMR were derived directly from the RG 1.232, appendix C, example PDCs for MHTGRs and are therefore identical to the RG PDCs or contain minor verbiage adaptations to reflect the MMR design. The NRC staff evaluated each proposed PDC for its applicability to the specific UIUC MMR design and evaluated whether the underlying safety bases for PDCs documented in RG 1.232 remained applicable for each in the context of the preliminary UIUC MMR design summarized earlier in this SE. The NRC staff finds that the following UIUC MMR PDCs are acceptable because they are applicable to the UIUC MMR design and because the underlying safety bases documented in RG 1.232 remain applicable and valid: 1, 2, 3, 4, 5, 10, 11, 12, 13, 14, 16, 18, 20, 21, 22, 23, 24, 25, 28, 29, 30, 31, 32, 34, 36, 37, 44, 45, 46, 60, 61, 62, 63, 64, and 72.

RG 1.232 Criteria Not Applicable to the UIUC MMR

The guidance in RG 1.232, appendix C, has criteria that are applicable to some advanced reactor designs other than MHTGRs, as included in the other appendices of the RG, but that are considered not applicable to MHTGR designs for various reasons. Many of these criteria are not considered applicable because of specific design aspects and features of MHTGRs; for example, some of these criteria are not applicable to MHTGRs because of their reliance on a functional containment in lieu of a traditional containment structure. Consistent with the RG 1.232, appendix C guidance related to MHTGRs such as the UIUC MMR, UIUC determined that the following RG 1.232 PDCs are not applicable to the UIUC MMR: 27, 33, 35, 38, 39, 40, 41, 42, 43, 50, 51, 52, 53, 54, 55, 56, and 57. The NRC staff reviewed each of these criteria and finds that exclusion of each is appropriate in the context of the associated RG 1.232 guidance and the UIUC MMR design because they are not applicable for MHTGRs, consistent with RG 1.232, appendix C.

UIUC MMR PDCs with Departures from RG 1.232

The NRC staff determined that the remaining proposed UIUC MMR PDCs warranted specific evaluation regarding their acceptability due to differences in verbiage that the NRC staff considered to be more-than-minimal departures from the underlying RG 1.232 verbiage on which they were based. These proposed UIUC MMR PDCs, 15, 17, 19, 26, 70a, 70b, 71a, and 71b, are discussed in the SE sections below.

UIUC MMR PDC 15 – Reactor Helium Pressure Boundary Design

UIUC MMR PDC 15 describes UIUC's proposed approach to reactor helium pressure boundary design. As discussed in TR appendix A, the language proposed deviates from the language in the analogous RG 1.232 MHTGR-DC 15. Specifically, UIUC added a reference to pressure relief valves to ensure the SSC's inclusion within the scope of the PDC. TR appendix A states that the verbiage was added because of conflicting definitions between the NRC and USNC terminology for the helium pressure boundary. TR appendix A furthermore states that a definition for the helium pressure boundary, as it applies to the UIUC MMR, will be provided in the planned UIUC MMR PSAR. UIUC's proposed definition for helium pressure boundary would be subject to NRC staff review, as appropriate, as part of its review of future licensing submittals. Nevertheless, the NRC staff finds that UIUC's addition of the specific SSC to ensure it is included in the scope of applicability of the PDC does not change the underlying intent of the PDC itself in any way, as the list of SSCs is exemplary in nature and not necessarily all-inclusive. The SSCs to which the PDC applies will be subject to NRC staff review, as appropriate, as part of its review of future licensing submittals. Therefore, the NRC staff finds that the underlying safety bases documented in RG 1.232 remain applicable. Accordingly, the NRC staff finds that proposed UIUC MMR PDC 15 is acceptable.

UIUC MMR PDC 17 – Electric Power Systems

UIUC MMR PDC 17 describes UIUC's proposed approach to electric power systems. As discussed in TR appendix A, the proposed wording deviates from that in the analogous RG 1.232 MHTGR-DC 17. Specifically, UIUC removed significant verbiage regarding electric power systems required to perform functions important to safety because the UIUC MMR will be designed to be passive and will not require electrical power for anticipated operational occurrences (AOOs) or postulated accidents. The remaining applicable verbiage is retained, largely unchanged, including a statement specifying that the design shall demonstrate that power for important to safety functions is provided. The NRC staff finds the proposed UIUC MMR PDC 17 acceptable on the basis that, in the context of the UIUC MMR research reactor design, including its passive nature, the proposed UIUC MMR PDC 17 aligns with the guidance provided in NUREG-1537, Parts 1 and 2, sections 8.1 and 8.2, regarding normal and emergency electrical power systems. Specifically, the guidance in NUREG-1537, Part 2, section 8.2, states that emergency electrical power is required when "assured power is required to maintain safe reactor shutdown … to support operation of a required engineered safety feature … or to protect the public from release of radioactive effluents." The NRC staff finds that UIUC MMR PDC 17meets the intent of this statement regarding power for important to safety functions. Specifically, the NRC staff expects this PDC to cover any potential important to safety functions that would require electrical power that fall outside the scope of AOOs or postulated accidents, if any are identified. In this context, the NRC staff finds that for the UIUC MMR, the proposed PDC 17 aligns with the underlying safety bases for MHTGR-DC 17, as documented in RG 1.232.

UIUC MMR PDC 19 – Control Room or Secure Remote Monitoring Facility

UIUC MMR PDC 19 describes UIUC's proposed approach to the UIUC MMR control room design and function and facility monitoring. As discussed in TR appendix A, the language proposed deviates from that in the analogous RG 1.232 MHTGR-DC 19. Specifically, the proposed verbiage reflects that no operator actions are required to maintain the reactor in a safe condition following an accident. Therefore, a control room will be provided to safely operate under normal conditions, but for accident conditions, only monitoring capability from a control room or remote monitoring facility is necessary. Additionally, in proposed UIUC MMR PDC 19, UIUC specified a more restrictive (lower) control room personnel radiation exposure limit of 2 rem, compared to 5 rem in RG 1.232 MHTGR-DC 19. The NRC staff finds that proposed UIUC MMR PDC 19 is acceptable because, considering that the UIUC MMR design does not require operator actions in response to accidents, the PDC aligns with the underlying safety bases for MHTGR-DC 19 as documented in RG 1.232 and also provides a conservative control room radiation exposure limit.

The NRC staff notes that its approval of this proposed UIUC MMR PDC does not constitute NRC staff approval of remote operations (e.g., operational control from a remote monitoring facility) of the UIUC MMR.

UIUC MMR PDC 26 – Reactivity Control Systems

UIUC MMR PDC 26 describes UIUC's proposed approach to the design of reactivity control systems and/or means. As discussed in TR appendix A, the proposed language deviates from that in the analogous RG 1.232 MHTGR-DC 26. Specifically, UIUC removed the verbiage "a minimum of two" in the text that precedes the four numbered paragraphs in the PDC. UIUC states that it made this change to remove potential ambiguity that may lead a reader to interpret the statement as suggesting that two reactivity control systems are needed for each of the numbered paragraphs. The NRC staff notes that, although it does not consider the intent of the wording in RG 1.232 MHTGR-DC 26 to require two reactivity control systems for each of the four numbered paragraphs, the remaining verbiage in proposed UIUC MMR PDC 26 continues to effectively require that two (or more) reactivity control systems or means be provided as part of the design to satisfy the entirety of the PDC, consistent with the intent of RG 1.232. The verbiage contained in the paragraphs numbered (1) through (3) is largely unchanged between RG 1.232 MHTGR-26 and proposed UIUC MMR PDC 26. In paragraph (4), UIUC removed "fuel loading" from the list of example intervention activities. However, the NRC staff finds that because the removed text is only an example activity, the removal is inconsequential to the underlying intent of the PDC. Based on the discussion above, the NRC staff finds that proposed UIUC MMR PDC 26 maintains the same effective underlying intent and basis for safety as RG 1.232 MHTGR-DC 26 and remains applicable to the UIUC MMR. Therefore, the NRC staff finds that proposed UIUC MMR PDC 26 is acceptable.

The NRC staff notes that its approval of proposed UIUC MMR PDC 26 with the "fuel loading" text removed does not constitute NRC staff approval of any specific UIUC MMR refueling technique, approach, or strategy.

UIUC MMR PDCs 70a and 70b – Reactor Vessel and Reactor System Structural Design Basis

The verbiage provided in RG 1.232 MHTGR-DC 70 contains two numbered requirements regarding the design of the reactor vessel and reactor system, to ensure integrity is maintained during postulated accidents to maintain (1) capability of passive heat removal of residual heat and (2) capability to insert control rods. As discussed in TR appendix A, proposed UIUC MMR PDCs 70a and 70b separate the two requirements of RG 1.232 MHTGR-DC 70 into two separate PDCs for organizational purposes. The NRC staff reviewed the proposed UIUC MMR PDCs 70a and 70b and finds that they adequately reflect the underlying intent of the RG 1.232 language and the underlying safety bases documented in RG 1.232 continue to appropriately apply to the UIUC MMR. Therefore, the NRC staff finds that the proposed UIUC MMR PDCs 70a and 70b are acceptable.

UIUC MMR PDCs 71a and 71b – Citadel Design Basis

As discussed in TR appendix A, similar to PDC 70, UIUC split the RG 1.232 MHTGR-DC 71 language into proposed UIUC MMR PDCs 71a and 71b for organizational purposes. Additionally, UIUC modified the language from the RG slightly to replace "reactor building" with "citadel" to reflect the MMR design terminology. The NRC staff reviewed the proposed UIUC MMR PDCs 71a and 71b and finds that they adequately reflect the underlying intent of the RG 1.232 language, and the underlying safety bases documented in RG 1.232 continue to appropriately apply to the UIUC MMR. Therefore, the NRC staff finds that the proposed UIUC MMR PDCs 71a and 71b are acceptable.

LIMITATIONS AND CONDITIONS

Approval of this TR for incorporation by reference to support licensing application(s) for construction and/or operation is limited to the MMR design located at the University of Illinois at Urbana Champaign, as the design is generally described in the subject topical report, licensed under the provisions of 10 CFR 50.21(c), as a class 104(c) research reactor. Any departure from the design, or licensing approach described in the TR would require justification in the referencing licensing action and would be subject to NRC review for acceptability.

CONCLUSION

The NRC staff concludes that the UIUC TR "University of Illinois Urbana-Champaign High-Temperature Gas-Cooled Research Reactor: Micro Modular Reactor (MMR™) Principal Design Criteria," Release 2 (Reference 2), section 4.0 and appendix A, describe an acceptable list of PDCs that can be used to meet the requirements of 10 CFR 50.34(a)(3)(i) in support of future licensing actions associated with the construction and operation of the proposed UIUC MMR. The NRC staff based this conclusion on the following findings:

- The proposed UIUC MMR technology is largely similar to that of the MHTGR described in the applicable regulatory guidance for generating MHTGR PDCs in RG 1.232, appendix C. The UIUC MMR PDCs were generated using, and closely align with, this guidance.
- Each UIUC MMR PDC is applicable to the UIUC MMR design and meets the underlying intent of the applicable RG 1.232 guidance, in the context of the specific UIUC MMR design.
- UIUC's determinations regarding RG 1.232 criteria that are not applicable to the UIUC MMR are appropriate and are consistent with the intent of the applicable RG 1.232 guidance, in the context of the UIUC MMR design.

REFERENCES

1. Letter from UIUC to NRC, "Submittal of 'University of Illinois Urbana-Champaign High Temperature Gas-cooled Research Reactor: Micro Modular Reactor (MMR™) Principal Design Criteria' Topical Report," dated November 15, 2023 (ML23319A407).

2. Letter from UIUC to NRC, "Submittal of 'University of Illinois Urbana-Champaign High Temperature Gas-cooled Research Reactor: Micro Modular Reactor (MMR™) Principal Design Criteria' Topical Report, Release 02 (Revision 1), dated April 10, 2024," dated April 10, 2024 (ML24101A413).

3. NRC Regulatory Guide 1.232, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors," Revision 0, dated April 2018 (ML17325A611).

4. Letter from UIUC to NRC, "USNRC Project No. 99902094: UIUC Regulatory Engagement Plan revision submission," dated June 26, 2023 (ML23178A259).

5. NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," Part 1, "Format and Content," and Part 2, "Standard Review Plan and Acceptance Criteria," dated February 1996 (ML042430055 and ML042430048).

6. SECY-18-0096, "Functional Containment Performance Criteria for Non-Light-Water-Reactors," dated September 28, 2018 (ML18114A546).

7. SRM-SECY-18-0096, "Staff Requirements – SECY-18-0096 – Functional Containment Performance Criteria for Non-Light-Water-Reactors," dated December 4, 2018 (ML18338A502).

Principal Contributors: D. Beacon, NRR
L. Kingsbury, NRR

Date: July 25, 2024

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CONFIGURATION CONTROL SUMMARY

Document Revision History

Document No.	Rel.	Date	Prepared By	Revision Description
IMRDD-MMR-23-06	01	11/10/2023	USNC	Initial Issue for NRC review
IMRDD-MMR-23-06	02	04/10/2024	USNC	Revision based upon corrections needed to resolve issues raised by the NRC during the Public Meeting on 04/05/2024. • Remove joint review request. • Updated PDC 15 to be consistent with RG 1.232 and to include pressure relief valves. • Updated PDC 16 to be consistent with RG 1.232. • Updated PDC 19 to be consistent with RG 1.232 • Updated PDC 32 to be consistent with RG 1.232. • Section 5.0 changed to be specific to the UIUC project.

DOCUMENT APPROVALS

Approvals	Name/Org.	Title	Signature	Date
Preparer	Michael Hamer USNC	Licensing Manager (acting)	DocuSigned by: Michael J Hamer B9D9582D1A9048D...	10-Apr-24
Approver	Michael Hamer for Zackary Rad USNC	Vice President, Regulatory Affairs & Quality	DocuSigned by: Michael J Hamer B9D9582D1A9048D...	10-Apr-24

Approvals	Name/Org.	Title	Signature	Date
Reviewer	Les Foyto UIUC	Director, Licensing	DocuSigned by: Les Foyto 2D40B8D068E64AF...	10-Apr-24
Approver	Caleb Brooks UIUC	Associate Professor & Project Lead	DocuSigned by: Caleb Brooks EB6137AE6E3C41A...	10-Apr-24

EXECUTIVE SUMMARY

This topical report (TR) summarizes the methodology for development of the principal design criteria (PDC) for the University of Illinois Urbana-Champaign (UIUC) Micro-Modular Reactor (MMR™) designed by the Ultra Safe Nuclear Corporation (USNC). The PDC were developed based on the key design features of the MMR™ technology and use of the modular high-temperature gas-cooled reactor design criteria (MHTGR-DC) provided in Regulatory Guide 1.232, "Guidance for Developing Principal Design Criteria for Advanced (Non-Light Water) Reactors." The resultant PDCs incorporate the key design features of the MMR™ technology for licensing of the MMR™ design for deployment at UIUC.

UIUC is requesting the U.S. Nuclear Regulatory Commission (NRC) to provide a safety evaluation of the proposed USNC MMR™ Principal Design Criteria described in Section 4 and listed Appendix A.

ACRONYMS & ABBREVIATIONS

This list contains the acronyms and abbreviations used in this document.

Abbreviation or Acronym	Definition
AC	Alternating Current
AGR	Advanced Gas Reactor
AOO	Anticipated Operational Occurrence
ARDC	Advanced Reactor Design Criteria
CNSC	Canadien Nuclear Safety Commission
CP	Construction Permit
ECCS	Emergency Core Cooling System
EU	Enriched Uranium
FCM®	Fully Ceramic Micro-encapsulated
GDC	General Design Criteria (10 CFR 50, Appendix A)
HALEU	High-Assay Low-Enriched Uranium (i.e., enriched 5% to 20% in ^{235}U, exclusive)
LEU	Low-Enriched Uranium (i.e., enriched 0.72% to 4.95% in ^{235}U)
LWR	Light Water Reactor
MOC	Memorandum of Cooperation
MHTGR	Modular High-Temperature Gas-cooled Reactor
MHTGR-DC	Modular High-Temperature Gas-cooled Reactor - Design Criteria
MMR™	Micro-Modular Reactor
NRC	[U.S.] Nuclear Regulatory Commission
PDC	Principal Design Criteria
RCCS	Reactor Cavity Cooling System
RCSS	Reactivity Control and Shutdown System
RG	Regulatory Guide
SARRDL	Specified Acceptable Radionuclide Release Design Limit
SiC	Silicon Carbide
SSC	Structures, Systems and Components
TR	Topical Report
TLDC	Top Level Design Criteria
TRISO	Tristructural Isotropic
UIUC	University of Illinois Urbana-Champaign
USNC	Ultra Safe Nuclear Corporation

1.0 INTRODUCTION

The Micro Modular Reactor (MMR™) design is being developed by Ultra Safe Nuclear Corporation (USNC) to be licensed and deployed as a non-power research reactor for the University of Illinois Urbana-Champaign (UIUC) in accordance with the applicable regulatory requirements of the U.S. Nuclear Regulatory Commission (NRC).

The Principal Design Criteria (PDC) provided in this report are based on the key design features of the USNC MMR technology, as summarized in Section 2, and the similarity to the MHTGR-DC design criteria that the NRC provides for advanced reactors in Regulatory Guide 1.232, Revision 0, Appendix C, dated April 2018. The demonstration that the MMR design satisfies these PDC will be provided in the UIUC license application documents (e.g., safety analysis reports, topical reports, etc.) required to be submitted by NRC regulations.

1.1. PURPOSE

This report provides the proposed PDCs for the MMR design developed by USNC using the guidance in RG 1.232 for review and approval by the NRC that will be deployed at UIUC.

1.2. SCOPE

NRC regulations in 10 CFR 50.34(a)(3)(i) require that applicants for a construction permit (CP) include the PDC for a proposed facility. USNC's proposed MMR PDCs are intended for use by UIUC for the non-power research reactor.

1.3. RELATIONSHIP TO OTHER DOCUMENTS

USNC has developed Top Level Design Criteria (TLDC) were referenced primarily from US NRC Regulatory Guide (RG) 1.232, Rev. 0. USNC then used the TLDC as the basis for the PDC that are applicable to the USNC MMR™ design.

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). The regulations note that the GDC are generally applicable to other types of reactor units and are intended to provide guidance in establishing the PDC for such other units. That is, the GDC in 10 CFR 50, Appendix A are guidance, not regulatory requirements, for non-LWRs. The NRC published RG 1.232, "Guidance for Developing Principal Design Criteria for Non-Light Water Reactors" (Ref. 2), dated April 2018, that provides guidance for establishing the PDC for non-light water reactor designs. RG 1.232 includes PDCs for the Modular High-Temperature Gas-Cooled Reactor (MHTGR). USNC has used the advanced reactor design criteria guidance provided by the NRC in RG 1.232 for MHTGRs to develop its proposed PDCs for the USNC MMR™ design for UIUC.

1.4. ACTION REQUESTED

UIUC requests an NRC safety evaluation of the USNC MMR™ PDCs provided in Section 4 and listed in Appendix A to be used for the UIUC MMR™.

1.5. DEFINITIONS

1.5.1 Defense in Depth (DiD): A hierarchical deployment of different levels of diverse equipment and procedures to prevent the escalation of anticipated operational occurrences and to maintain the effectiveness of physical barriers placed between a radiation source or radioactive material and workers, members of the public or the environment, in operational states and, for some barriers, in accident conditions.

1.5.2 Functional Containment: A barrier or set of barriers taken together, that effectively limit the physical transport and release of radionuclides to the environment across a full range of normal operating conditions, AOEs, and accident conditions.

1.5.3 Non-Safety Related (NSR): Relating to SSCs, items, and human actions that are not classified as safety related.

1.5.4 Safety Function: A specific purpose that must be accomplished for safety for a facility or activity to prevent or to mitigate radiological consequences of normal operation, anticipated operational occurrences and accident conditions.

Note: This guidance is commonly condensed into a succinct expression of three fundamental safety functions for nuclear power plants: (a) Control of reactivity; (b) Cooling of radioactive material; (c) Confinement of radioactive material.

1.5.5 Safety Related (SR): Relating to SSCs, items, and human actions that have an impact on safety, and are relied upon to remain functional during and following design basis events to ensure the three fundamental safety functions as defined in 1.5.4 are satisfied. The three fundamental safety functions are used in lieu of the functions for safety related SSCs in 10 CFR 50.2.

2.0 DESIGN FEATURES

The MRR is a modular high temperature gas cooled reactor that uses proprietary Fully Ceramic Micro-Encapsulated (FCM®) pellets that are stacked in columns in solid hexagonal graphite blocks. The FCM pellets incorporate fuel comprised of tristructural isotropic (TRISO) particles embedded in silicon carbide (SiC). The MMR uses an inert gas (helium) as the heat transfer medium. The MMR is designed with a passive safety response to accidents and relies on functional containment as the primary means to limit release of radioactivity to the environment.

The MMR design uses technology and safety capabilities considerably different from the Light Water Reactor (LWR) technology that is the focus of many of the NRC regulations. For example, the MMR does not require an active or passive emergency core cooling system (ECCS) to rapidly replenish primary coolant to recover the fuel in the event of a rupture of the primary pressure boundary. Large safety margins are provided by both the fuel and the reactor design.

- The fuel is comprised of TRISO particles, which provide a highly effective fission product retention capability. The superior fission product retention capability of TRISO fuel particles enables the concept of “functional containment” in which these particles serve as the first containment barrier when operated within the range of qualification parameters.

- The TRISO particles in MMR fuel are encased in an FCM pellet of SiC that provides an additional layer of defense-in-depth for the retention of fission products by functional containment.
- The low power density of the active fuel region leads to slow fuel heat-up during loss of heat removal events.
- Low thermal power results in a small inventory available for release of the most limiting short-lived fission products for public safety, such as ^{131}I and ^{85}Kr. The increased inventory of long-lived fission products associated with a long core life is addressed by the defense-in-depth approach to functional containment.
- The low power rating also reduces the decay heat that must be removed in postulated accidents, simplifying passive decay heat removal.
- Heat transfer fluid used for core cooling during normal operation is an inert, chemically stable, single-phase gas (helium) at operating pressures less than those of LWRs.
- Safety-related core cooling is passive and capable of maintaining fuel and component temperatures below limits with no helium, electrical power, or operator action.
- Secondary heat transfer is performed by a molten salt loop that effectively isolates the reactor from transients in the adjacent plant power conversion system.
- The reactor is located below grade. Although it does not have nor need a leak-tested containment building, it is surrounded by a concrete structure (the citadel) that serves as a barrier to release of radioactivity to the environment and provides protection against external hazards.

Table 2-1 below provides a comparison of the MMR design features to those of the other MHTGR designs that differ from LWRs.

Table 2-1 Comparison of MMR design features with MHTGR design

Design Feature	MHTGR	MMR
Core Design	Fully ceramic TRISO pebble fuel	Fully ceramic micro-encapsulated TRISO particle fuel
Power density	Low	Low
Heat transport (coolant)	Helium gas	Helium gas
Coolant activity	Low (graphite/air)	Low (graphite/air)
Moderator	Graphite	Graphite
Operating Pressure	High	High (but lower than LWRs)
Fission product mobility following fuel damage	Low (retained in fuel)	Low (retained in fuel)
Containment	N/A (based on functional containment concept)	N/A (based on functional containment concept)
Decay Heat Removal	Passive	Passive
Intermediate loop between primary and power conversion	No	Yes (intermediate heat exchanger with molten salt loop)

3.0 MMR PDC DEVELOPMENT METHODOLOGY

This section describes the process used by USNC to develop the PDC for the MMR design to be constructed at UIUC.

The MMR PDC development process began with a review of the advanced reactor design criteria (ARDC) from RG 1.232, Appendix C, to determine the relevance of the MHTGR-DC to the key design features of the MMR technology. Each ARDC in Appendix C of RG 1.232 was reviewed for applicability to the MMR design, considering the underlying safety basis for the ARDC and the supporting information in Appendix C of RG 1.232. In some cases, the ARDC in RG 1.232 adopts the GDC from 10 CFR 50, Appendix A without change.

Based on the similarities with many design features of MHTGR technologies, the MHTGR-DC were specifically reviewed for relevancy to the MMR design to determine whether the MHTGR-DC in RG 1.232, Appendix C, should be considered for inclusion in the proposed PDC for the MMR design. When the USNC review of the MHTGR-DC concluded that those specific ARDC could be directly adopted as written for the MMR design, the MHTGR-DC were selected as the PDC for the MMR design.

For those MHTGR-DC that did not fully apply to the key design features of the MMR design, the MHTGR-DC were assessed to determine if changes to the MHTGR-DC could be made such that they were representative of the MMR design without compromising safety. This assessment is based on technical relevance and the amount of modification that would be necessary to conform to the MHTGR-DC to be representative of the MMR design. Modifications were made to reflect the design of the MMR and the departures from the underlying criteria are annotated (underlined/strikethrough) during development of the proposed MMR PDCs. In most cases, the MHTGR-DC provided in Appendix C to RG 1.232 did not need to be changed for applicability to the MMR design and were adopted as written.

Once the complete set of proposed MMR PDC were developed, a methodical review was performed to ensure that the PDC collectively provide a comprehensive design and regulatory framework for the MMR design. This was done by evaluating each of the major unique design attributes of the MMR design and comparing it against the set of proposed PDC to ensure that there is a PDC that captures the attribute or topic.

The RG 1.232 review method described above was performed by USNC personnel knowledgeable in the MMR technology and development of the MMR design. The results of the review are documented along with a basis for selection of the MMR PDC. The results of the review and the proposed MMR PDC are provided in Section 4. The MMR PDC were internally reviewed by USNC engineering and licensing personnel.

For the MMR standard design, the proposed MMR PDC are intended to be applicable to all SSCs identified as important to safety. However, the MMR at UIUC is intended to be licensed as a non-power research reactor. Consequently, the safety classification methodology for the UIUC application only includes two SSC classifications: safety related, and non-safety related. Therefore, all references of “important to safety” SSCs and functions in the proposed PDC in Section 4.0 and Appendix A should be interpreted as SSCs and functions classified as “safety-related” for the UIUC MMR. The term important to safety is retained in this TR to maintain consistency with MMR standard design documents.

4.0 PROPOSED MMR PDC

The proposed MMR PDCs are subdivided into categories similar to those used for the MHTGR-DC in RG 1.232, Appendix C as follows:

I. Overall Requirements
II. Multiple Barriers
III. Reactivity Control
IV. Heat Transport Systems
V. Reactor Containment
VI. Fuel and Radioactivity Control
VII. Additional MHTGR-DC

Proposed changes to RG 1.232 MHTGR design criteria to accommodate the USNC MMR design to be deployed at UIUC are documented in the table provided in Appendix A. Changes are identified by strikeouts to indicate language removed from the MHTGR-DC and underlines are provided for language additions to the MMR PDC. Example provided below.

This PDC indicates that USNC has replaced the term *reactor building* with *citadel*.

EXAMPLE:

PDC No.	NRC RG 1.232, Appendix C MHTGR-DC Title and Content	USNC MMR PDC	Comments
72	Provisions for periodic ~~reactor building~~ inspection. The ~~reactor building~~ shall be designed to permit (1) appropriate periodic inspection of all important structural areas and the depressurization pathway, and (2) an appropriate surveillance program.	Provisions for periodic citadel inspection. The <u>citadel</u> shall be designed to permit (1) appropriate periodic inspection of all important structural areas and the depressurization pathway, and (2) an appropriate surveillance program.	Minor deviations from RG 1.232. Consistent with the intent of MHTGR-DC.

Consistent with the ARDCs provided for MHTGRs in RG 1.232, the proposed PDCs for the MMR design do not include PDCs associated with a containment structure. As discussed in Appendix C to RG 1.232, the GDCs associated with the containment structure are very specific to a pressure retaining containment structure and include containment design basis, fracture prevention of containment pressure boundary, capability for containment leakage rate testing, provisions for containment testing and inspection, piping systems penetrating containment, reactor coolant boundary penetrating containment, containment isolation, and closed system isolation valves. As noted in Appendix C to RG 1.232, the MHTGR designs do not have a pressure retaining reactor containment structure. The USNC MMR relies upon a multi-barrier functional containment fuel pellet to control the release of radionuclides. On that basis, the eight criteria associated with the containment structure were deemed to be not applicable to the MHTGR-DC. Based on the similarity of the key design features between the MMR and MHTGR designs, the MHTGR-DC approach of not including PDCs associated with pressure retaining containment structures is applicable to the MMR design.

A comparison of the proposed MMR PDCs with the advanced reactor MHTGR-DC in RG 1.232, Appendix C, is provided as Attachment A to this report. This comparison provides notes for differences between the USNC MMR PDCs and the MHTGR-DC with a suitable justification for necessary changes.

4.1 OVERALL REQUIREMENTS

4.1.1. Quality standards and records (PDC-1)

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.

4.1.2. Design bases for protection against natural phenomena (PDC-2)

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.

4.1.3. Fire protection (PDC-3)

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.

4.1.4. Environmental and dynamic effects design bases (PDC-4)

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 Regulator demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

4.1.5 Sharing of structures, systems, and components (PDC-5)

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.

4.2 MULTIPLE BARRIERS

4.2.1. Reactor design (PDC-10)

The reactor system and associated heat removal, control, and protection systems shall be designed with appropriate margin to ensure that specified acceptable system radionuclide release design limits (SARRDL) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

4.2.2. Reactor inherent protection (PDC-11)

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.

4.2.3. Suppression of reactor power oscillations (PDC-12)

The reactor core and associated control and protection systems shall be designed to ensure that power oscillations that can result in conditions exceeding specified acceptable system radionuclide release design limits (SARRDL) are not possible or can be reliably and readily detected and suppressed.

4.2.4. Instrumentation and control (PDC-13)

Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions, as appropriate, to ensure adequate safety, including those variables and systems that can affect the fission process and the integrity of the reactor core, reactor helium pressure boundary, and functional containment. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

4.2.5. Reactor helium pressure boundary (PDC-14)

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.

4.2.6. Reactor helium pressure boundary design (PDC-15)

All the systems that are part of the helium pressure boundary, such as the reactor system, vessel system, and heat removal systems, including the pressure relief valves 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.

4.2.7. Containment design (PDC-16)

A reactor functional containment, consisting of multiple barriers internal and/or external to the reactor and its cooling system, shall be provided to control the release of radioactivity to the environment and ensure the functional containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

4.2.8. Electric power systems (PDC-17)

With electrical power not needed for anticipated operational occurrences or postulated accidents, the design shall demonstrate that power for important to safety functions is provided.

Electrical power is not needed for anticipated operational occurrences or postulated accidents. The design shall demonstrate that power for important to safety functions is provided.

4.2.9. Inspection and testing of electric power systems (PDC-18)

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.

4.2.10. Control room or secure remote monitoring facility (PDC-19)

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and either a control room or a remote monitoring facility shall be provided to monitor it 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 2 rem (20 mSv) total effective dose equivalent for the duration of the accident.

The design shall ensure that operator action is not required to maintain the nuclear power unit in a safe condition under accident conditions.

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.

4.2.11 Citadel design basis - Release during Depressurization Accident (PDC-71b).

The design of the citadel shall be such that, during postulated accidents, it structurally protects the geometry to provide a pathway for the release of reactor helium from the building in the event of depressurization accidents.

4.2.12. Provisions for periodic Citadel inspection (PDC-72)

The Citadel shall be designed to permit (1) appropriate periodic inspection of all important structural areas and the depressurization pathway, and (2) an appropriate surveillance program.

4.3 REACTIVITY CONTROL

4.3.1. Protection system functions (PDC-20)

The protection system shall be designed (1) to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that the specified acceptable system radionuclide release design limit (SARRDL) is not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

4.3.2. Protection system reliability and testability (PDC-21)

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.

4.3.3. Protection system independence (PDC-22)

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.

4.3.4. Protection system failure modes (PDC-23)

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.

4.3.5. Separation of protection and control systems (PDC-24)

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.

4.3.6. Protection system requirements for reactivity control malfunctions (PDC-25)

The protection system shall be designed to ensure that specified acceptable system radionuclide release design limits (SARRDL) are not exceeded during any anticipated operational occurrence, accounting for a single malfunction of the reactivity control systems.

4.3.7. Reactivity control systems (PDC-26)

The reactivity control systems or means shall provide: a) A means of inserting negative reactivity at a sufficient rate and amount to assure, with appropriate margin for malfunctions, that the specified acceptable system radionuclide release design limits (SARRDL) and the reactor helium pressure boundary design limits are not exceeded, and safe shutdown is achieved and maintained during normal operation, including anticipated operational occurrences. b) A means which is independent and diverse from the other(s), shall be capable of controlling the rate of reactivity changes resulting from planned, normal power changes to assure that the SARRDL and the reactor helium pressure boundary design limits are not exceeded. c) 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. d) A means for holding the reactor shutdown under conditions which allow for interventions such as inspection and repair shall be provided.

4.3.8. Reactivity limits (PDC-28)

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.

4.3.9. Protection against anticipated operational occurrences (PDC-29)

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.

4.3.10. Reactor vessel and reactor system structural design basis (PDC-70b)

The design of the reactor vessel and reactor system shall be such that their integrity is maintained during postulated accidents to permit sufficient insertion of the neutron absorbers to provide for reactor shutdown.

4.4 HEAT TRANSPORT SYSTEMS

4.4.1. Quality of reactor helium pressure boundary integrity including reducing fluid ingress (PDC-30)

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.

4.4.2. Fracture prevention of reactor helium pressure boundary (PDC-31)

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 helium composition, including contaminants and reaction products, on material properties, (3) residual, steady-state, and transient stresses, and (4) size of flaws.

4.4.3. Inspection of reactor helium pressure boundary (PDC-32)

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.

4.4.4. Passive residual heat removal (PDC-34)

A passive system to remove residual heat shall be provided for normal operations and anticipated operational occurrences, the system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core to an ultimate heat sink at a rate such that specified acceptable system radionuclide release design limits (SARRDL) and the design conditions of the reactor helium pressure boundary are not exceeded. During postulated accidents, the system safety function shall provide effective cooling. Suitable redundancy in components and features and suitable interconnections, leak detection, and isolation capabilities shall be provided to ensure the system safety function can be accomplished, assuming a single failure.

4.4.5. Inspection of passive residual heat removal system (PDC-36)

The passive residual heat removal system shall be designed to permit appropriate periodic inspection of important components to ensure the integrity and capability of the system.

4.4.6. Testing of passive residual heat removal system (PDC-37)

The passive residual heat removal system shall be designed to permit appropriate periodic functional testing to ensure (1) the structural and leak tight integrity of its components <u>as applicable</u>, (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.

4.4.7. Structural and equipment cooling (PDC-44)

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.

4.4.8. Inspection of structural and equipment cooling systems (PDC-45)

The structural and equipment cooling systems shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the systems.

4.4.9. Testing of structural and equipment cooling systems (PDC-46)

The structural and equipment cooling systems shall be designed to permit appropriate periodic functional testing to assure (1) the structural and leak tight integrity of their components, (2) the operability and the performance of the system components, and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequences that bring the systems into operation for reactor shutdown and postulated accidents, including operation of associated systems.

4.4.10. Citadel design basis (PDC-70a)

The design of the reactor vessel and reactor system shall be such that their integrity is maintained during postulated accidents to ensure the geometry for passive removal of residual heat from the reactor core to the ultimate heat sink.

4.4.11. Citadel design basis – Residual Heat Removal (PDC-71a)

The design of the Citadel shall be such that, during postulated accidents, it structurally protects the geometry for passive removal of residual heat from the reactor core to the ultimate heat sink.

4.5 FUEL AND RADIOACTIVITY CONTROL

4.5.1. Control of releases of radioactive materials to the environment (PDC-60)

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.

4.5.2. Fuel storage and handling and Radioactivity control (PDC-61)

The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment confinement, and filtering systems. 4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage cooling under accident conditions.

4.5.3. Prevention of criticality in fuel storage and handling (PDC-62)

Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

4.5.4. Monitoring fuel and waste storage (PDC-63)

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.

4.5.5. Monitoring radioactivity releases (PDC-64)

Means shall be provided for monitoring the Citadel atmosphere, effluent discharge paths, and plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

5.0 CONCLUSIONS AND RECOMMENDATIONS

USNC performed a comprehensive review of the advanced reactor design criteria provided by the NRC in RG 1.232, Appendix C, for modular high-temperature gas-cooled reactors and developed a set of proposed principal design criteria (PDC) for its MMR design based on the MHGTR-DC that meet the requirements of 10 CFR 50.34(a)(3)(i) and the underlying safety objectives of 10 CFR 50 Appendix A as applicable. In addition, the proposed PDCs meet the requirements of 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).

These proposed PDC reflect the key features of the MMR design and provide an appropriate set of requirements to facilitate the licensing of the MMR design. As such, once approved, these PDC apply for use by UIUC for the USNC MMR design under 10 CFR 50.

6.0 REFERENCES

6.1 U.S. Nuclear Regulatory Commission, Regulatory Guide 1.232, “Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors,” Rev. 0, dated April 2018.

6.2 USNC “MMR Top Level Design Criteria Specification”, Release 02, dated 06 October 2023.

7.0 APPENDICES

7.1 **APPENDIX A:** Comparison of RG 1.232 Appendix C MHTGR DC and USNC MMR PDC

PDC No.	NRC RG 1.232, Appendix C MHTGR-DC Title and Content	USNC MMR PDC	Comment
	I. OVERALL REQUIREMENTS		
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.	**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.	No deviations from RG 1.232. Consistent with MHTGR-DC.
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	**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	No deviations from RG 1.232. Consistent with MHTGR-DC.

PDC No.	NRC RG 1.232, Appendix C MHTGR-DC Title and Content	USNC MMR PDC	Comment
	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.	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.	
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.	**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.	No deviations from RG 1.232. Consistent with MHTGR-DC.
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	**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	No deviations from RG 1.232. Consistent with MHTGR-DC.

PDC No.	NRC RG 1.232, Appendix C MHTGR-DC Title and Content	USNC MMR PDC	Comment
	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.	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 Regulator demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.	
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.	**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.	No deviations from RG 1.232. Consistent with MHTGR-DC. The UIUC research reactor deployment does not currently include plans for more than one unit which will be noted in the SAR.

PDC No.	NRC RG 1.232, Appendix C MHTGR-DC Title and Content	USNC MMR PDC	Comment
	II. MULTIPLE BARRIERS		
10	**Reactor design.** The reactor system and associated heat removal, control, and protection systems shall be designed with appropriate margin to ensure that specified acceptable system radionuclide release design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.	**Reactor design.** The reactor system and associated heat removal, control, and protection systems shall be designed with appropriate margin to ensure that specified acceptable system radionuclide release design limits (SARRDL) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.	Added acronym (SARRDL). No material deviations from RG 1.232. Consistent with MHTGR-DC.
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.	**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.	No deviations from RG 1.232. Consistent with MHTGR-DC.
12	**Suppression of reactor power oscillations.** The reactor core and associated control and protection systems shall be designed to ensure that power oscillations that can result in conditions exceeding specified acceptable system radionuclide release design limits are not possible or can be reliably and readily detected and suppressed.	**Suppression of reactor power oscillations**. The reactor core and associated control and protection systems shall be designed to ensure that power oscillations that can result in conditions exceeding specified acceptable system radionuclide release design limits (SARRDL) are not possible or can be reliably and readily detected and suppressed.	Added acronym (SARRDL) No material deviations from RG 1.232. Consistent with MHTGR-DC.
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	**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	No deviations from RG 1.232. Consistent with MHTGR-DC.

PDC No.	NRC RG 1.232, Appendix C MHTGR-DC Title and Content	USNC MMR PDC	Comment
	fission process and the integrity of the reactor core, reactor helium pressure boundary, and functional containment. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.	fission process and the integrity of the reactor core, reactor helium pressure boundary, and functional containment. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.	
14	**Reactor helium pressure boundary.** The reactor helium pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, of gross rupture, and of unacceptable ingress of moisture, air, secondary coolant, or other fluids.	**Reactor 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.	No deviations from RG 1.232. Consistent with MHTGR-DC.
15	**Reactor helium pressure boundary design.** All systems that are part of the reactor helium pressure boundary, such as the reactor system, vessel system, and heat removal systems, and the associated auxiliary, control, and protection systems, shall be designed with sufficient margin to ensure that the design conditions of the reactor helium pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.	**Reactor helium pressure boundary design.** All systems that are part of the reactor helium pressure boundary, such as the reactor system, vessel system, and heat removal systems, <u>including the pressure relief valve</u>s 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.	Minor deviation from RG 1.232 to add "…including the pressure relief valves …" because of conflicting definitions between the NRC and USNC terminology for the helium pressure boundary, to be described in the UIUC PSAR.
16	**Containment design.** A reactor functional containment, consisting of multiple barriers internal and/or external to the reactor and its cooling system, shall be provided to control the release of radioactivity to the environment and to ensure that the functional containment design conditions important to safety	**Containment design.** A reactor functional containment, consisting of multiple barriers internal and/or external to the reactor and its cooling system, shall be provided to control the release of radioactivity to the environment and ensure the functional containment design conditions important to safety are not	No deviations from RG 1.232. Consistent with MHTGR-DC.

PDC No.	NRC RG 1.232, Appendix C MHTGR-DC Title and Content	USNC MMR PDC	Comment
	are not exceeded for as long as postulated accident conditions require.	exceeded for as long as postulated accident conditions require.	
17	**Electric power systems.** ~~Electric power systems shall be provided when required to permit functioning of structures, systems, and components. The safety function for each power system shall be to provide sufficient capacity and capability to ensure that (1) that the specified acceptable system radionuclide release design limits and the reactor helium pressure boundary design limits are not exceeded as a result of anticipated operational occurrences and (2) safety functions that rely on electric power are maintained in the event of postulated accidents.~~ ~~The electric power systems shall include an onsite power system and an additional power system. The onsite electric power system shall have sufficient independence, redundancy, and testability to perform its safety functions, assuming a single failure. An additional power system shall have sufficient independence and testability to perform its safety function.~~ ~~If~~ electric power is not needed for anticipated operational occurrences or postulated accidents, the design shall demonstrate that power for important to safety functions is provided.	**Electric power systems.** Electrical power is not needed for anticipated operational occurrences or postulated accidents. The design shall demonstrate that power for important to safety functions is provided.	Deviations from the language used in RG 1.232. The requirement for a specific additional power system is not needed for the MMR design due to its passive nature and the requirement for redundancy already included for the onsite electric power system. The proposed PDC is considered to be functionally consistent with this MHTGR-DC.
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	**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	No deviations from RG 1.232. Consistent with MHTGR-DC.

PDC No.	NRC RG 1.232, Appendix C MHTGR-DC Title and Content	USNC MMR PDC	Comment
	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.	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.	
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.	**Control room <u>or secure remote monitoring facility.</u>** A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and <u>either a control room or a remote monitoring facility shall be provided to monitor it</u> 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 <u>2 rem (20 mSv)</u> total effective dose equivalent for the duration of the accident. <u>The design shall ensure that operator action is not required to maintain the nuclear power unit in a safe condition under accident conditions.</u> 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	Minor deviations from RG 1.232 due the passive cooling nature of the MMR design, no operator actions are credited nor required to maintain the reactor in a safe condition following an accident, therefore, the term “monitor” is used instead of “maintain”. Changed radiation exposure level from 5 rem to 2 rem to align with conservative dose limits, recognized

PDC No.	NRC RG 1.232, Appendix C MHTGR-DC Title and Content	USNC MMR PDC	Comment
		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.	internationally that the MMR design is capable of achieving due to the significantly lower source term of typical LWRs.
71b	**~~Reactor building~~ design basis. [MHTGR-DC-71]** The design of the ~~reactor building~~ shall be such that, during postulated accidents, it structurally protects the geometry ~~for passive removal of~~ ~~residual heat from the reactor core to the ultimate~~ ~~heat sink and~~ provide~~s~~ a pathway for the release of reactor helium from the building in the event of depressurization accidents.	**Citadel design basis - Release during Depressurization Accident.** The design of the citadel shall be such that, during postulated accidents, it structurally protects the geometry to provide a pathway for the release of reactor helium from the building in the event of depressurization accidents.	MHTGR-DC-71 was separated and categorized resulting in two distinctive design criteria for functional clarity (71a & 71b).
72	**Provisions for periodic ~~reactor building~~ inspection.** The ~~reactor building~~ shall be designed to permit (1) appropriate periodic inspection of all important structural areas and the depressurization pathway, and (2) an appropriate surveillance program.	**Provisions for periodic citadel inspection.** The citadel shall be designed to permit (1) appropriate periodic inspection of all important structural areas and the depressurization pathway, and (2) an appropriate surveillance program.	Minor deviations from RG 1.232. Consistent with the intent of MHTGR-DC.

PDC No.	NRC RG 1.232, Appendix C MHTGR-DC Title and Content	USNC MMR PDC	Comment
	III. REACTIVITY CONTROL		
20	**Protection system functions.** The protection system shall be designed (1) to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that the specified acceptable system radionuclide release design limits is not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.	**Protection system functions.** The protection system shall be designed (1) to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that the specified acceptable system radionuclide release design limits (<u>SARRDL</u>) is not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.	Added acronym (SARRDL) Minor deviations from RG 1.232. Consistent with the intent of MHTGR-DC.
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.	**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.	No deviations from RG 1.232. Consistent with MHTGR-DC.

PDC No.	NRC RG 1.232, Appendix C MHTGR-DC Title and Content	USNC MMR PDC	Comment
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.	**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.	No deviations from RG 1.232. Consistent with MHTGR-DC.
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.	**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.	No deviations from RG 1.232. Consistent with MHTGR-DC.

PDC No.	NRC RG 1.232, Appendix C MHTGR-DC Title and Content	USNC MMR PDC	Comment
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.	**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.	No deviations from RG 1.232. Consistent with MHTGR-DC.
25	**Protection system requirements for reactivity control malfunctions.** The protection system shall be designed to ensure that specified acceptable system radionuclide release design limits are not exceeded during any anticipated operational occurrence, accounting for a single malfunction of the reactivity control systems.	**Protection system requirements for reactivity control malfunctions.** The protection system shall be designed to ensure that specified acceptable system radionuclide release design limits (SARRDL) are not exceeded during any anticipated operational occurrence, accounting for a single malfunction of the reactivity control systems.	Added acronym (SARRDL) No material deviations from RG 1.232. Consistent with the intent of MHTGR-DC.

PDC No.	NRC RG 1.232, Appendix C MHTGR-DC Title and Content	USNC MMR PDC	Comment
26	**Reactivity control systems.** ~~A minimum of two~~ reactivity control systems or means shall provide: (1) A means of inserting negative reactivity at a sufficient rate and amount to assure, with appropriate margin for malfunctions, that the specified acceptable system radionuclide release design limits and the reactor helium pressure boundary design limits are not exceeded and safe shutdown is achieved and maintained during normal operation, including anticipated operational occurrences. (2) A means which is independent and diverse from the other(s), shall be capable of controlling the rate of reactivity changes resulting from planned, normal power changes to assure that the specified acceptable system radionuclide release design limits and the reactor helium pressure boundary design limits are not exceeded. (3) A means of inserting negative reactivity at a sufficient rate and amount to assure, with appropriate margin for malfunctions, that the capability to cool the core is maintained and a means of shutting down the reactor and maintaining, at a minimum, a safe shutdown condition following a postulated accident. (4) A means for holding the reactor shutdown under conditions which allow for interventions such as ~~fuel loading~~, inspection and repair shall be provided.	**Reactivity control systems.** The reactivity control systems or means shall provide: (1) A means of inserting negative reactivity at a sufficient rate and amount to assure, with appropriate margin for malfunctions, that the specified acceptable system radionuclide release design limits (SARRDL) and the reactor helium pressure boundary design limits are not exceeded, and safe shutdown is achieved and maintained during normal operation, including anticipated operational occurrences. (2) A means which is independent and diverse from the other(s), shall be capable of controlling the rate of reactivity changes resulting from planned, normal power changes to assure that the SARRDL and the reactor helium pressure boundary design limits are not exceeded. (3) A means of inserting negative reactivity at a sufficient rate and amount to assure, with appropriate margin for malfunctions, that the capability to cool the core is maintained and a means of shutting down the reactor and maintaining, at a minimum, a safe shutdown condition following a postulated accident. (4) A means for holding the reactor shutdown under conditions which allow for interventions such as inspection and repair shall be provided.	Minor deviations from the language used in RG 1.232. Otherwise, consistent with MHTGR-DC. Deleted: "A minimum of two" and replaced with "The". The wording in the original text was suggesting that there must be two of each for the four sections mentioned. USNC interpretation is that two systems must satisfy at least one of the four sections, and not all.

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27	**Combined reactivity control systems capability.** DELETED—Information incorporated into MHTGR-DC 26	**N/A**	No deviations from RG 1.232. Consistent with MHTGR-DC.
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.	**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.	No deviations from RG 1.232. Consistent with MHTGR-DC.
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.	**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.	No deviations from RG 1.232. Consistent with MHTGR-DC.

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70b	**Reactor vessel and reactor system structural design basis. [MHTGR-DC-70]** The design of the reactor vessel and reactor system shall be such that their integrity is maintained during postulated accidents ~~(1) to ensure the geometry for passive removal of residual heat from the reactor core to the ultimate heat sink and (2)~~ to permit sufficient insertion of the neutron absorbers to provide for reactor shutdown.	**Reactor vessel and reactor system structural design basis.** The design of the reactor vessel and reactor system shall be such that their integrity is maintained during postulated accidents to permit sufficient insertion of the neutron absorbers to provide for reactor shutdown.	MHTGR-DC-70 items 1 and 2 are separated and categorized resulting in two distinctive design criteria for functional clarity as 70a & 70b.

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	IV. HEAT TRANSPORT SYSTEMS		
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.	**Quality of reactor helium pressure boundary integrity including reducing fluid ingress.** 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.	No material deviations from RG 1.232. Consistent with MHTGR-DC. Title changed for clarity.
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 helium composition, including contaminants and reaction products, on material properties, (3) residual, steady-state, and transient stresses, and (4) size of flaws.	**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 helium composition, including contaminants and reaction products, on material properties, (3) residual, steady-state, and transient stresses, and (4) size of flaws.	No deviations from RG 1.232. Consistent with MHTGR-DC.

PDC No.	NRC RG 1.232, Appendix C MHTGR-DC Title and Content	USNC MMR PDC	Comment
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.	**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.	No deviations from RG 1.232. Consistent with MHTGR-DC.
33	**Reactor coolant makeup.** Not applicable to MHTGR.	**N/A**	No deviations from RG 1.232.
34	**Passive residual heat removal.** A passive system to remove residual heat shall be provided. For normal operations and anticipated operational occurrences, the system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core to an ultimate heat sink at a rate such that specified acceptable system radionuclide release design limits and the design conditions of the reactor helium pressure boundary are not exceeded. During postulated accidents, the system safety function shall provide effective cooling. Suitable redundancy in components and features and suitable interconnections, leak detection, and isolation capabilities shall be provided to ensure the system safety function can be accomplished, assuming a single failure.	**Passive (source range) residual heat removal.** A passive system to remove residual heat shall be provided. For normal operations and anticipated operational occurrences, the system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core to an ultimate heat sink at a rate such that specified acceptable system radionuclide release design limits (SARRDL) and the design conditions of the reactor helium pressure boundary are not exceeded. During postulated accidents, the system safety function shall provide effective cooling. Suitable redundancy in components and features and suitable interconnections, leak detection, and isolation capabilities shall be provided to ensure the system safety function can be accomplished, assuming a single failure.	Added acronym (SARRDL) No material deviations from RG 1.232. Consistent with MHTGR-DC.
35	**Emergency core cooling.** Not applicable to MHTGR.	**N/A**	No deviations from RG 1.232.

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36	**Inspection of passive residual heat removal system.** The passive residual heat removal system shall be designed to permit appropriate periodic inspection of important components to ensure the integrity and capability of the system.	**Inspection of passive residual heat removal system.** The passive residual heat removal system shall be designed to permit appropriate periodic inspection of important components to ensure the integrity and capability of the system.	No deviations from RG 1.232. Consistent with MHTGR-DC
37	**Testing of passive residual heat removal system.** The passive residual heat removal system shall be designed to permit appropriate periodic functional testing to ensure (1) the structural and 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.	**Testing of passive residual heat removal system.** The passive residual heat removal system shall be designed to permit appropriate periodic functional testing to ensure (1) the structural and 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.	No deviations from RG 1.232 Consistent with MHTGR-DC.
38	**Containment heat removal.** Not applicable to MHTGR.	**N/A**	No deviations from RG 1.232.
39	**Inspection of containment heat removal system.** Not applicable to MHTGR.	**N/A**	No deviations from RG 1.232.
40	**Testing of containment heat removal system.** Not applicable to MHTGR.	**N/A**	No deviations from RG 1.232.

PDC No.	NRC RG 1.232, Appendix C MHTGR-DC Title and Content	USNC MMR PDC	Comment
41	**Containment atmosphere cleanup.** Not applicable to MHTGR.	**N/A**	No deviations from RG 1.232.
42	**Inspection of containment atmosphere cleanup systems.** Not applicable to MHTGR.	**N/A**	No deviations from RG 1.232.
43	**Testing of containment atmosphere cleanup systems.** Not applicable to MHTGR.	**N/A**	No deviations from RG 1.232.
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.	**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.	No deviations from RG 1.232 Consistent with MHTGR-DC.
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 assure the integrity and	**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 assure the integrity and	No deviations from RG 1.232. Consistent with MHTGR-DC.

PDC No.	NRC RG 1.232, Appendix C MHTGR-DC Title and Content	USNC MMR PDC	Comment
	capability of the systems.	capability of the systems.	
46	**Testing of structural and equipment cooling systems.** The structural and equipment cooling systems shall be designed to permit appropriate periodic functional testing to assure (1) the structural and leak tight integrity of their components, (2) the operability and the performance of the system components, and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequences that bring the systems into operation for reactor shutdown and postulated accidents, including operation of associated systems.	**Testing of structural and equipment cooling systems.** The structural and equipment cooling systems shall be designed to permit appropriate periodic functional testing to assure (1) the structural and leak tight integrity of their components, (2) the operability and the performance of the system components, and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequences that bring the systems into operation for reactor shutdown and postulated accidents, including operation of associated systems.	No deviations from RG 1.232. Consistent with MHTGR-DC.
70a	**Reactor vessel and reactor system structural design basis. [MHTGR-DC-70]** The design of the reactor vessel and reactor system shall be such that their integrity is maintained during postulated accidents (1) to ensure the geometry for passive removal of residual heat from the reactor core to the ultimate heat sink and (2) to permit sufficient insertion of the neutron absorbers to provide for reactor shutdown.	**Reactor vessel and reactor system structural design basis.** The design of the reactor vessel and reactor system shall be such that their integrity is maintained during postulated accidents to ensure the geometry for passive removal of residual heat from the reactor core to the ultimate heat sink.	MHTGR-DC-70 items (1) & (2) are separated and re-categorized resulting in two distinctive design criteria for functional clarity as PDCs 70a & 70b with no changes to RG 1.232 MHTGR-DC-70, item (1).

PDC No.	NRC RG 1.232, Appendix C MHTGR-DC Title and Content	USNC MMR PDC	Comment
71a	**Reactor building design basis. [MHTGR-DC-71]** The design of the reactor building shall be such that, during postulated accidents, it structurally protects the geometry for passive removal of residual heat from the reactor core to the ultimate heat sink. and provides a pathway for the release of reactor helium from the building in the event of depressurization accidents.	**Citadel design basis – Residual Heat Removal.** The design of the Citadel shall be such that, during postulated accidents, it structurally protects the geometry for passive removal of residual heat from the reactor core to the ultimate heat sink.	MHTGR-DC-71 was separated and categorized resulting in two distinctive design criteria for functional clarity (71a & 71b).

PDC No.	NRC RG 1.232, Appendix C MHTGR-DC Title and Content	USNC MMR PDC	Comment
V. REACTOR CONTAINMENT			
50	**Containment design basis.** Not applicable to MHTGR.	**N/A**	No deviations from RG 1.232.
51	**Fracture prevention of containment pressure boundary.** Not applicable to MHTGR.	**N/A**	No deviations from RG 1.232.
52	**Capability for containment leakage rate testing.** Not applicable to MHTGR.	**N/A**	No deviations from RG 1.232.
53	**Provisions for containment testing and inspection.** Not applicable to MHTGR.	**N/A**	No deviations from RG 1.232.
55	**Reactor coolant boundary penetrating containment.** Not applicable to MHTGR.	**N/A**	No deviations from RG 1.232.
56	**Primary Containment isolation.** Not applicable to MHTGR.	**N/A**	No deviations from RG 1.232.
57	**Closed system isolation valves.** Not applicable to MHTGR.	**N/A**	No deviations from RG 1.232.
54	**Piping systems penetrating containment.** Not applicable to MHTGR.	**N/A**	No deviations from RG 1.232.

PDC No.	NRC RG 1.232, Appendix C MHTGR-DC Title and Content	USNC MMR PDC	Comment
	VI. FUEL AND RADIOACTIVITY CONTROL		
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.	**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.	No deviations from RG 1.232. Consistent with MHTGR-DC.
61	**Fuel storage and handling and radioactivity control.** The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage cooling under accident conditions.	**Fuel storage and handling and radioactivity control.** The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage cooling under accident conditions.	No deviations from RG 1.232. Consistent with MHTGR-DC

PDC No.	NRC RG 1.232, Appendix C MHTGR-DC Title and Content	USNC MMR PDC	Comment
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.	**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.	No deviations from RG 1.232. Consistent with MHTGR-DC.
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.	**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.	No deviations from RG 1.232. Consistent with MHTGR-DC.
64	**Monitoring radioactivity releases.** Means shall be provided for monitoring the ~~reactor building~~ atmosphere, effluent discharge paths, and plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.	**Monitoring radioactivity releases.** Means shall be provided for monitoring the <u>Citadel</u> atmosphere, effluent discharge paths, and plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.	Minor deviation from RG 1.232 to replace “reactor building” with “Citadel”, otherwise, consistent with MHTGR-DC.

PDC No.	NRC RG 1.232, Appendix C MHTGR-DC Title and Content	USNC MMR PDC	Comment
VII. Additional MHTGR-DC			
70	**Reactor vessel and reactor system structural design basis.** The design of the reactor vessel and reactor system shall be such that their integrity is maintained during postulated accidents (1) to ensure the geometry for passive removal of residual heat from the reactor core to the ultimate heat sink and (2) to permit sufficient insertion of the neutron absorbers to provide for reactor shutdown.	**Reactor vessel and reactor system structural design basis.** See PDC-70a and 70b.	MHTGR-DC-70 was separated and categorized resulting in two distinctive design criteria for functional clarity (70a & 70b).
71	**Reactor building design basis.** The design of the reactor building shall be such that, during postulated accidents, it structurally protects the geometry for passive removal of residual heat from the reactor core to the ultimate heat sink and provides a pathway for the release of reactor helium from the building in the event of depressurization accidents.	**Reactor building design basis.** See PDC-71a and 71b.	MHTGR-DC-71 was separated and categorized resulting in two distinctive design criteria for functional clarity (71a & 71b).
72	**Provisions for periodic reactor building inspection.** The reactor building shall be designed to permit (1) appropriate periodic inspection of all important structural areas and the depressurization pathway, and (2) an appropriate surveillance program.	**Provisions for periodic reactor building inspection.** See comment	MHTGR-DC-72 is provided in Section 4.2 and Appendix A, Category II as a “Multiple Barrier”