Kairos Power

KP-NRC-2006-001

June 12, 2020

Project No. 99902069

US Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Subject: Kairos Power LLC
Topical Report Submittal
Principal Design Criteria for the Kairos Power Fluoride Salt-Cooled, High Temperature Reactor

References:

1. Kairos Power LLC, "Principal Design Criteria for the Kairos Power Fluoride Salt-Cooled, High Temperature Reactor," Revision 0, December 21, 2018 (ML19087A103)
2. Kairos Power LLC, "Principal Design Criteria for the Kairos Power Fluoride Salt-Cooled, High Temperature Reactor," Revision 1, July 31, 2019 (ML19184A238)
3. Nuclear Regulatory Commission, Letter from Benjamin Beasley to Peter Hastings, "Safety Evaluation for Kairos Power LLC Topical Report Principal Design Criteria for the Kairos Power Fluoride Salt-Cooled, High Temperature Reactor (Revision 1)," May 22, 2020 (ML20111A118)

Kairos Power submitted Revision 0 of the subject topical report for Nuclear Regulatory Commission (NRC) review on December 21, 2018 (Reference 1). Revision 1 of the Kairos Power Principal Design Criteria topical report was submitted on July 31, 2019 (Reference 2). On May 22, 2020 the NRC transmitted the final safety evaluation of this topical report (Reference 3).

The Enclosure to this report provides the approved version of the report, designated as KP-TR-003-NP-A. The approved version also includes the May 22, 2020 letter from the NRC and its final safety evaluation report. The prior versions of this report included proprietary information but Kairos Power has since elected to make this report non-proprietary in its their entirety.

If you have any questions or need any additional information, please contact James Tomkins at tomkins@kairospower.com or (805) 215-6129, or Darrell Gardner at gardner@kairospower.com or (704) 769-1226.

Sincerely,

Peter Hastings, PE
Vice President, Regulatory Affairs and Quality

Enclosure:

1) Principal Design Criteria for the Kairos Power Fluoride Salt-Cooled, High Temperature Reactor

xc (w/enclosure):

Benjamin Beasley, Chief, Advanced Reactor Licensing Branch
Stewart Magruder, Project Manager, Advanced Reactor Licensing Branch

Enclosure 1

Principal Design Criteria for the Kairos Power Fluoride Salt-Cooled, High Temperature Reactor

(Non-Proprietary)

CONTENTS

Section A

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

May 22, 2020

Mr. Peter Hastings
Vice President, Regulatory Affairs
and Quality
Kairos Power LLC
707 W Tower Ave
Alameda, CA 94501

SUBJECT: SAFETY EVALUATION FOR KAIROS POWER LLC TOPICAL REPORT "PRINCIPAL DESIGN CRITERIA FOR THE KAIROS POWER FLUORIDE SALT-COOLED, HIGH TEMPERATURE REACTOR" (REVISION 1) (CAC NO. 000431)

Dear Mr. Hastings:

By letter dated December 21, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18355B067), Kairos Power LLC (Kairos) submitted for U.S. Nuclear Regulatory Commission (NRC) staff review, "Principal Design Criteria for the Kairos Power Fluoride Salt-Cooled, High Temperature Reactor Topical Report." The NRC staff sent a preliminary set of questions to Kairos on May 23, 2019 (ADAMS Accession No. ML19144A291). By letter dated July 31, 2019, Kairos submitted "Principal Design Criteria for the Kairos Power Fluoride Salt-Cooled, High Temperature Reactor" (Revision 1) (ADAMS Accession No. ML19212A756).

The NRC staff documented its review in the enclosed safety evaluation (SE) which was previously provided to you for comments on March 4, 2020 (ADAMS Accession No. ML20015A502). You provided comments to the NRC staff on April 1, 2020 (ADAMS Accession No. ML20094M049). The NRC staff has incorporated your comments, as appropriate, in the enclosed SE. The enclosed SE is final and will be made publicly available as specified in your comments.

If you have any questions, please contact Stewart Magruder at 301-348-5766 or by email at Stewart.Magruder@nrc.com

Sincerely,

/RA/

Benjamin Beasley, Chief
Advanced Reactor Licensing Branch
Division of Advanced Reactors and Non-Power
Production and Utilization Facilities
Office of Nuclear Reactor Regulation

Project No. 99902069

Enclosure:
Final Safety Evaluation

SUBJECT: SAFETY EVALUATION FOR KAIROS POWER LLC TOPICAL REPORT "PRINCIPAL DESIGN CRITERIA FOR THE KAIROS POWER FLUORIDE SALT-COOLED, HIGH TEMPERATURE REACTOR" (REVISION 1) (CAC NO. 000431) DATED MAY 22, 2020

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ADAMS Accession No.: ML20111A118 ***via e-mail**

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DATE	5/04 /2020	5/22/2020	

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SAFETY EVALUATION

"PRINCIPAL DESIGN CRITERIA FOR THE KAIROS POWER FLUORIDE SALT-COOLED,

HIGH TEMPERATURE REACTOR" (REVISION 1)

PROJECT NO. 99902069

1.0 INTRODUCTION

By letter dated December 21, 2018, Kairos Power LLC (Kairos Power, the applicant) submitted for the U.S. Nuclear Regulatory Commission (NRC) staff's review, "Principal Design Criteria for the Kairos Power Fluoride Salt-Cooled, High Temperature Reactor Topical Report" (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18355B067). On March 6, 2019, the NRC staff found that the material presented in the topical report (TR) provides the technical information in sufficient detail to enable the NRC staff to conduct a detailed technical review (ADAMS Accession No. ML19059A355).

Kairos Power requested the NRC staff's review and approval of its proposed principal design criteria (PDCs), which are to be used by applicants of the Kairos Power Fluoride Salt-Cooled, High Temperature Reactor (KP-FHR) design for future licensing submittals under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 or Part 52. As part of the NRC staff's review, initial feedback and questions were provided to the applicant on May 23, 2019 (ADAMS Accession No. ML19144A315). In response to these questions and a teleconference between the NRC staff and Kairos Power, the applicant submitted Revision 1 of the TR on July 31, 2019 (ADAMS Package Accession No. ML19212A755). This safety evaluation (SE) is based on Revision 1 of the TR.

2.0 REGULATORY EVALUATION

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

Specifically, the following Commission regulations pertain to the PDCs:

- 10 CFR 50.34(a)(3)(i), which requires, in part, that applications for a CP include PDCs for the facility. An OL would reference a CP, which would include PDCs.

- 10 CFR 52.47(a)(3)(i), which requires, in part, that applications for a DC include PDCs for the facility.

- 10 CFR 52.79(a)(4)(i), which requires, in part, that applications for a COL include PDCs for the facility.

- 10 CFR 52.137(a)(3)(i), which requires, in part, that applications for a SDA include PDCs for the facility.

- 10 CFR 52.157(a), which requires, in part, that applications for a ML include PDCs for the reactor to be manufactured.

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

Recognizing that the GDCs in 10 CFR Part 50, Appendix A may not be appropriate for non-light-water reactors (non-LWRs), the NRC issued Regulatory Guide (RG) 1.232, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors," (Reference 1) which serves as guidance to develop PDCs for non-LWR designs.

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

3.0 TECHNICAL EVALUATION

3.1 INTRODUCTION

To support future licensing actions regarding the KP-FHR under 10 CFR Part 50 or Part 52, the applicant submitted the TR to engage with the NRC staff regarding the development of its design-specific PDCs. The applicant noted that the GDC in 10 CFR Part 50, Appendix A, function as guidance, not regulatory requirements, for non-LWRs. The applicant therefore used RG 1.232 to develop PDCs for the design. Guidance in RG 1.232 provides a general set of advanced reactor design criteria (ARDCs), and also includes design criteria for two specific non-LWR designs, the Sodium-Cooled Fast Reactor (SFR) and the Modular High-Temperature Gas-Cooled Reactor (MHTGR). The TR evaluates how the ARDC, the SFR-specific design criteria (SFR-DC), and the MHTGR-specific design criteria (MHTGR-DC), apply to the KP-FHR design and concludes that they collectively provide a comprehensive design and regulatory framework for the KP-FHR design.

The TR provides the PDCs developed by the applicant for the KP-FHR. The PDCs developed for the KP-FHR are informed by the guidance in RG 1.232 and take into consideration attributes unique to the KP-FHR design. The primary purpose of the TR, as stated by the applicant, is to request "NRC review and approval of these PDCs to be used by applicants of the KP-FHR design for standard DCs, COLs, SDAs, and MLs under the applicable regulations in 10 CFR 52; or limited work authorizations, CPs and OLs under 10 CFR 50." The applicant further stated that the license application documents (e.g., safety analysis reports) required to be submitted by the cited regulations will demonstrate that the KP-FHR design satisfies these PDCs.

3.1.1 Design Features

Section 1.1 of the TR provides an overview of the key design features of the KP-FHR. For a contextual comparison with other non-LWR designs, the applicant provides Table 1, "Comparison of Advanced Reactor Designs." The applicant stated, "that the KP-FHR contains design features similar in nature to those found in the SFR or MHTGR, and it does not add fundamentally new or unique features from those present in the SFR or MHTGR designs."

The KP-FHR uses tri-structural isotropic (TRISO) particles in pebble form as fuel, and fluoride salt to cool the reactor. The applicant stated that the coolant is maintained at "near-atmospheric pressure" and circulated via pumps. The primary coolant transfers heat to an intermediate heat exchanger loop with nitrate salt that is "compatible with reactor coolant." The applicant also stated that the design uses a normal decay heat removal system and a natural circulation vessel cooling decay heat removal system.

Rather than a traditional containment building, the KP-FHR utilizes a functional containment approach, consistent with SECY-18-0096 (Reference 4), "Functional Containment Performance Criteria for Non-Light-Water-Reactors" and the associated SRM-SECY-18-0096 "Staff Requirements - SECY-18-0096 - Functional Containment Performance Criteria for Non-Light Water-Reactors" (Reference 5). The applicant stated that the ultimate design objective of the functional containment is to meet offsite dose requirements at the plant's exclusion area boundary with margin. The TRISO fuel particles are the first and primary barrier against the release of radionuclides. The fluoride coolant is also capable of retaining fission products, aiding in ensuring radionuclides are not released beyond applicable limits.

3.1.2 Regulatory Review

Section 1.2 of the TR outlines the applicant's discussion of the applicable regulations. The regulations under 10 CFR Parts 50 and 52 require that all applicants for a CP, SDA, COL, SDA, or ML provide PDCs. For LWRs, the GDCs set forth in 10 CFR Part 50, Appendix A provide the minimum requirements for the PDCs. The applicant noted that while the GDCs have served as a key part of the regulatory framework for LWRs, they are only generally applicable to other types of reactor units and are intended to provide guidance in establishing the PDCs.

The ARDCs in RG 1.232 were informed by the GDCs and provide guidelines for PDCs for non-LWR designs. The ARDCs are intended to be technology inclusive, and the RG provides technology-specific design criteria for the SFR and the MHTGR. The applicant chose to apply both the technology-inclusive ARDCs and technology-specific criteria as applicable, because the KP-FHR has design elements similar to those used in developing the SFR-DCs and MHTGR-DCs.

The applicant references relevant portions of the RG, including noting that "in each case, it is the responsibility of the designer or applicant to provide not only the PDCs for the design but also supporting information that justifies to the NRC how the design meets the PDCs submitted, and how the PDCs demonstrate adequate assurance of safety." Together with a comparison to RG 1.232, justification is provided for each of the PDCs proposed in the TR.

3.1.3 NRC Staff Evaluation

In reviewing the KP-FHR design, the NRC staff identified several key design features that influenced the development of the PDCs. These features include:

- A chemically stable coolant. Adverse interactions between the coolant/fuel, coolant/coolant boundary, and coolant/atmosphere all represent important considerations that could merit their own PDCs, similar to those specified in the SFR-DC

70-series in RG 1.232. While the applicant stated that the coolant is "chemically stable," the applicant has not demonstrated this feature at this stage of review. Verification of the coolant performance will be necessary to ensure that the proposed PDCs related to the reactor coolant and reactor coolant system represent an adequate set of criteria.

- TRISO fuel particles and fuel pebbles. This fuel form represents the foundation of the functional containment approach proposed by the applicant. The NRC staff noted that the applicant will still need to establish and document performance criteria consistent with the methodology outlined in SECY-18-0096. This entails identifying: event sequences to ensure the plant-level performance criteria are met, those SSCs and programmatic controls needed to fulfill important safety functions, and controlling parameters for the design and operation of risk-significant SSCs. The applicant stated that the TRISO fuel particles and the reactor coolant provide the credited functions during accident conditions and that the integrity of the entire reactor coolant system is not necessary during accident conditions. If additional design features are needed to provide credited design functions, the adequacy of these PDCs should be re-evaluated.

- An intermediate coolant loop using a coolant that is chemically compatible with reactor coolant. A compatible intermediate system coolant, which precludes interactions with the reactor coolant boundary and the reactor coolant obviates the need for a PDC for the intermediate system.

- "Near-atmospheric" pressure for the reactor coolant system. The absence of an energetic release of coolant during loss-of-coolant type accident results in a fundamentally different risk profile of the KP-FHR compared to LWR designs.

The applicant is requesting approval for the proposed PDCs without the detailed system specifications, drawings, and calculations. Continued development of the KP-FHR may result in changes to design features outlined in Section 1 of the TR. In this event, a revision to the proposed PDCs described in the TR may be necessary. These key design features of the KP-FHR, if changed, could necessitate the modification or addition of PDCs, and therefore, the NRC staff restricts the use of the TR as discussed in Section 4.0 of this SE.

As stated in the regulatory evaluation section of this SE, an applicant for a CP, SDA, COL, SDA, or ML under 10 CFR Part 50 or Part 52 is required to include PDCs for the facility. The applicant elected to use RG 1.232 to develop its PDCs. The NRC staff views this process as acceptable to establish the PDCs for the design and notes that the TR makes no finding on how any future submittal will demonstrate how the PDCs are satisfied for the design. The applicant has not decided on a licensing process to use and has requested the PDCs defined in this SE to be applicable to the different 10 CFR Part 50 and Part 52 licensing requirements pertaining to PDCs.

The scope of a future potential ML referencing 10 CFR 52.137 for a KP-FHR is not clear at this stage, nor would the NRC staff expect it to be, as this is not an application for an ML. The applicability of PDCs to a proposed ML may or may not cover the full extent of the PDCs for the complete design, but a subsequent CP, COL, or referenced DC would be required to address this discrepancy, if any. As such, the applicability of the TR for a future potential ML is conditional on referencing and interfacing with another license application (DC, CP, or COL) that would cover the full scope of the design. This is stated in Section 4.0 of this SE.

3.2 PDC DEVELOPMENT METHODOLOGY

Section 2 of the TR describes the process used by the applicant to develop the PDCs for the KP-FHR. The applicant relied on the ARDCs, SFR-DCs, and MHTGR-DCs as specified in

RG 1.232 to develop its design-specific PDCs. The KP-FHR does not directly parallel the design-specific ARDCs in RG 1.232, but it does share some features with the designs used to develop the ARDCs. The applicant chose to assess the ARDCs and adapt them to the KP-FHR, as applicable. The applicant used a review process to evaluate each of the KP-FHR design attributes against the PDCs that included evaluation by the applicant's engineering and licensing staff along with industry experts to ensure that the PDCs adequately capture each attribute. Figure 1, "Flow Chart of PDC Development Methodology," of the TR shows the process that the applicant followed to develop the PDCs.

3.2.1 NRC Staff Evaluation

The NRC staff finds that the TR properly utilizes the PDC development process described in RG 1.232. When the applicant deviates from the verbatim guidance in RG 1.232, a rationale is provided describing how the changes relate to the safety basis of the KP-FHR. Similarly, when the applicant elects to not utilize a PDC in RG 1.232, the applicant includes a justification from a safety perspective for the omission. Applicants and licensees may voluntarily use the guidance in RG 1.232 to demonstrate compliance with the underlying NRC regulations regarding PDCs. As stated in RG 1.232, methods or solutions that differ from those described in RG 1.232 may be deemed acceptable if a sufficient basis and supporting information is provided for the NRC staff to verify that the proposed alternative demonstrates compliance with the appropriate NRC regulations. The evaluation of the PDCs is provided in the subsequent sections of this SE.

3.3 RESULTS

3.3.1 Summary of Results

Section 3.1 of the TR provides a summary of the results and applicability of RG 1.232 to the KP-FHR. A brief discussion of each of the design criteria sections is provided:

Section I—Overall Requirements (Criteria 1–5) – general non-design specific requirements.

Section II—Multiple Barriers (Criteria 10–19) – criteria tailored to the barriers and systems used to control and contain the release of radioactivity; some notable changes proposed by the applicant.

Section III—Reactivity Control (Criteria 20–29) – criteria related to protection and reactivity control; the criteria are not particularly design-specific.

Section IV—Fluid Systems/Heat Transport Systems (Criteria 30–46) – criteria related to fluid, coolant, and heat transfer systems; these tend to be very design-specific and are revised for the KP-FHR design.

Section V—Reactor Containment (Criteria 50–57) – criteria related to pressure retaining containment; because the applicant has proposed a functional containment approach, these design criteria were deemed not to be applicable by the applicant, consistent with the MHTGR-DC approach.

Section VI—Fuel and Radioactivity Control (Criteria 60–64) – criteria related to radioactive releases, fuel and waste storage and handling; because of the broad-spectrum of unique fuel types considered in formulating RG 1.232, some changes were proposed by the applicant.

Section VII has two parts:

- Additional SFR-DC (Criteria 70–77) – criteria specific to SFRs, generally related to coolant purity and the use of an intermediate loop; the applicant considered these criteria and chose those applicable to the KP-FHR.

- Additional MHTGR-DC (Criteria 70–72) – criteria specific to MHTGRs, generally related to the reactor vessel and reactor building; the applicant considered these criteria and chose those applicable to the KP-FHR.

Appendices A and B of the TR provide a detailed description of the PDCs proposed by the applicant, including a basis for incorporating, changing, or not adopting the design criteria listed in RG 1.232.

3.3.2 Summary of Changes to the ARDC, SFR-DC, and MHTGR-DC

In many cases, the applicant stated that the design criteria apply as written; in other cases, the applicant stated that the proposed PDCs were changed to accommodate specific aspects of the KP-FHR design. The PDCs that were amended include those pertaining to fuel design limits, containment, and the coolant boundary in the context of the KP-FHR. The applicant also revised the PDCs to address language associated with the term “important to safety” as used in the GDC and RG 1.232, terminology associated with shutdown, and terms that are not applicable to the KP-FHR design. Other changes to the design criteria were made to accommodate the specific design of the KP-FHR, and all the changes are discussed in further detail in Section 3.3.3 of this SE.

3.3.2.1 NRC Staff Evaluation

The applicant has stated in the TR that it plans to use the guidance in DG-1353, “Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors,”(Reference 3) and NEI 18-04, “Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development,” (Reference 2) to develop its application and inform the safety classification of SSCs for the design. Due to the nature of the methodology used to develop the PDCs and inform the design, there are portions of the review that would be difficult to carry out at the current stage, as the implementation of the methodology is an iterative process that requires a full accounting of the design to assess risk- and safety-importance of plant SSCs. As such, the NRC staff added a condition on the use of the methodology discussed in the TR, referencing DG-1353 and NEI 18-04. This is discussed in Section 4.0 of this SE.

3.3.3 Detailed KP-FHR Results

Appendices A and B of the TR present the proposed PDCs for the KP-FHR. Section 3.3.1 of the TR provides a summary of the format and organization of the PDCs as presented.

3.3.3.1 NRC Staff Evaluation

Due to the nature of the KP-FHR design, which combines features envisioned in the development of both the MHTGR-DCs and SFR-DCs, there is not a single set of design criteria from RG 1.232 that can be used as a baseline to develop design-specific PDCs for the KP-FHR design. Instead, the applicant chose to select primarily from the generic ARDC and MHTGR-DC, supplementing with design criteria from the SFR-DC where appropriate.

PDC with No Changes

In the case of the following proposed PDCs, the applicant proposed using either the associated ARDC or MHTGR-DC from RG 1.232 with no changes for use as KP-FHR PDCs: 11, 21, 22, 23, 24, 25, 29, 36, 37, 45, 46, 60, 62, 63, 64, and 74. The NRC staff agrees that these PDCs are sufficiently broad to apply to the KP-FHR, and the rationale for the underlying safety basis documented in RG 1.232 remains applicable. As such, the NRC staff finds these PDCs to be acceptable.

PDC with Single Change - Using Safety Significant Instead of Important to Safety

A further set of proposed PDCs for KP-FHR, including 1, 2, 3, 4, 15, 18, 20, 28, 44 and 61 from RG 1.232, are unchanged from the design criteria in RG 1.232 with one exception: use of the term "safety significant" instead of the term "important to safety." Other proposed PDCs, including 5, 16, 17, 71, and 73 from RG 1.232, also make this change, but have more substantive changes and are thus discussed further below. As stated in 10 CFR Part 50, Appendix A, SSCs that are classified as "important to safety" are those "that provide reasonable assurance that the nuclear power plant can be operated without undue risk to the health and safety of the public." This definition could be applicable, but since the applicant plans to use the guidance in NEI 18-04 to the extent that it is endorsed by the NRC staff in DG-1353, the applicant has proposed to use the term "safety significant" where "important to safety" was present in the design criteria.

In NEI 18-04, safety significant SSCs are those classified as safety-related, those that perform a risk-significant function, and those that are needed to meet defense-in-depth criteria. NEI 18-04 provides context for what would cause an SSC to be classified as safety-related within the bounds of that methodology. For the purposes of this SE, the NRC staff would augment the definition of "safety-related" in NEI 18-04 to account for the regulatory definition of the term contained in 10 CFR 50.2. That is, for an applicant or licensee referencing the TR, SSCs that meet the definition of safety-related in 10 CFR 50.2 as applicable to the design would also fall within the scope of safety significant SSCs. Coupled with the limitation related to use of the guidance in DG-1353 and NEI-18-04 (that it is conditional on the NRC staff's approval of the implementation by the applicant of the guidance in DG-1353), the NRC staff finds this definition to be acceptable in that it appropriately defines the set of SSCs that would "provide reasonable assurance that the nuclear power plant can be operated without undue risk to the health and safety of the public." This is discussed in Section 4.0 of this SE.

Aside from the above change in wording, these PDCs are identical to those in RG 1.232. Similar to the previously referenced PDCs, the reasoning for the use of those design criteria, were described in RG 1.232. The NRC staff agrees that these PDCs are sufficiently broad to apply to the KP-FHR, and that the rationale for the underlying safety basis documented in RG 1.232 remains applicable. As such, the NRC staff finds these PDCs to be acceptable.

PDC Related to Functional Containment

The following set of ARDC from RG 1.232 were omitted by the applicant from its proposed PDCs: 38, 39, 40, 41, 42, 43, 50, 51, 52, 53, 54, 55, 56 and 57. The applicant proposed to use a functional containment approach, as described in SECY-18-0096. The applicant's PDC 16 is based on RG 1.232 MHTGR-DC 16 and is expected to fulfill the intended role of the ARDC listed as omitted above.

As discussed in MHTGR-DC 16, in Appendix C of RG 1.232, the term "functional containment" can be defined as "a set of barriers taken together, that effectively limit the physical transport and release of radionuclides to the environment across a full range of normal operating

conditions, AOOs [anticipated operational occurrences], and accident conditions." As described in the TR, the applicant "relies primarily on the multiple barriers within the TRISO fuel particles and fuel pebble" while also crediting the salt coolant as a "distinct barrier providing retention of fission products that escape the fuel particle and fuel pebble barriers." In general terms, use of this approach is acceptable to the NRC staff; however, this SE makes no finding on the acceptability of the functional containment design objective, performance requirements, and/or performance criteria that will be used to demonstrate the adequacy of this approach to meet regulatory requirements and provide reasonable assurance of public health and safety. The NRC staff expects that the establishment of these performance criteria and how they will demonstrate regulatory requirements are met to be the subject of future licensing submittals. With these limitations, the NRC staff finds that the applicant's proposed PDC 16 and related omission of the aforementioned ARDC to be acceptable.

Additional PDC

More substantial changes to the ARDC, SFR-DC, and MHTGR-DC (and/or related bases) were made to the following PDCs in RG 1.232: 5, 10, 12, 13, 14, 17, 19, 26, 30, 31, 32, 33, 34, 35, 70, 71, 73, 75, and 76. The NRC staff's evaluations of the applicant's proposed PDCs follow.

The applicant's proposed PDCs 5 and 19 change the terminology associated with shutdown (specifically "hot" or "cold" shutdown). Historically, PDCs associated with shutdown include specific temperature requirements. For the KP-FHR, adding specific temperature requirements to shutdown terminology is not appropriate, as the KP-FHR uses a salt mixture coolant that has phase change conditions substantially different than water. As such, the motivation behind requiring temperature conditional shutdown conditions is no longer applicable. Instead of "hot" or "cold" shutdown, the applicant proposed to adopt the term "safe shutdown," consistent with the discussion in SECY-94-084 (Reference 6), requiring adequate reactor subcriticality, decay heat removal, and radioactive material containment, for PDCs referencing shutdown. As such, the use of "the ability to achieve and maintain safe shutdown" in place of "an orderly shutdown and cooldown" in PDC 5 is appropriate and meets the underlying purpose and safety basis documented for ARDC 5 in RG 1.232 and is therefore, acceptable. The other change to PDC 5 involves the use of "safety significant" as discussed above. Based on the above discussion, the NRC staff finds the applicant's proposed PDC 5 to be acceptable. Similar changes to PDC 19 to remove the modifier "hot" from shutdown and replace "cold" with "safe" shutdown are also acceptable for the same reasons.

The applicant's proposed PDC 19 maintains the language proposed in ARDC 19 used to ensure that the control room supports operator actions as required during both normal and accident conditions. Therefore, the NRC staff finds this treatment to be acceptable with respect to human factors considerations. Proposed PDC 19 also maintains language from ARDC 19 that is used to ensure that the control room design provides adequate radiation protection to permit access and occupancy of the control room as required under accident conditions. Therefore, the NRC staff finds this treatment to be acceptable with respect to consideration of radiation protection in the control room design. Based on the above discussion, the NRC staff finds the applicant's proposed PDC 19 to be acceptable.

The applicant's proposed PDC 10 makes only a single change compared to MHTGR-DC 10: the phrase "reactor system" is replaced with "reactor core." Because the KP-FHR uses TRISO fuel particles like the MHTGR envisioned in the development of the MHTGR-DC, use of the MHTGR-DC is appropriate. The applicant stated the use of "core" over "system" when referring to the reactor region will distinguish between the contributing sources of dose – in the generic MHTGR envisioned in RG 1.232, circulating dose could be released from anywhere in the reactor coolant pressure boundary. In the KP-FHR, due to key design features including the TRISO fuel, "near-atmospheric" primary coolant pressures, and the ability to ensure core

cooling by maintaining coverage of the fuel with reactor coolant, the applicant stated that the change in terms is appropriate because dose limits are met using specified acceptable system radionuclide release design limits imposed on SSCs in the core region. These key design features are included as part of the limitations in this SE and discussed in Section 4.0. This evaluation makes no findings on how the dose limits are achieved, only that proposed PDC 10 provides an acceptable foundational design criterion that conforms with the rationale for MHTGR-DC 10 and provides appropriate requirements for the reactor core.

In PDCs 12, 17, 26, and 33 from RG 1.232, language changes are made to accommodate the fuel form. Specifically, as compared with the ARDC, "system radionuclide release" replaces "fuel" in the applicant's proposed PDCs 12, 33 and 34, and "specified acceptable system radionuclide release design limits" replaces "design limits for the fission product barriers" in the applicant's proposed PDCs 17 and 26. The KP-FHR proposes a functional containment approach using TRISO fuel particles and fuel pebbles. Combined with other design features, this approach acts to restrict radionuclide releases. Referring only to the fuel, rather than the entire system, would not be appropriate. Therefore, the use of fuel performance terms related to traditional fuel designs would not be appropriate for TRISO fuel. Design limits related to the fuel and related radionuclide retention systems will need to be developed by an applicant referencing the TR such that all pertinent regulatory fuel and dose requirements, including those in 10 CFR Parts 20, 50, 51, and 52 are met, as applicable for the specific application. This SE makes no finding on how a proposed design would meet those requirements. However, provided an applicant referencing the TR can demonstrate compliance with the pertinent regulatory fuel and dose requirements, this approach is consistent with the intended purpose and safety basis documented in RG 1.232.

The applicant's proposed PDC 12 makes only the change documented above when compared with ARDC 12 and is therefore, acceptable for the reasons discussed above. In addition to the changes outlined related to fuel, the applicant's proposed PDC 17 also makes the change to "safety significant" outlined above. Those changes are acceptable subject to the limitations discussed earlier in this section. Aside from those changes, the rationale for the applicant's proposed PDC 17, conforms with the rationale in RG 1.232 and is therefore, acceptable.

The applicant's proposed PDC 26 changes only the language referenced above but is discussed separately here as the applicant chose to omit PDC 27, in accordance with the position laid out in RG 1.232. As stated in RG 1.232, ARDC 26 combines the scope of GDC 26 and GDC 27. The development of ARDC 26 was informed by the proposed historical general design criteria, current GDC 26 and 27, the definition of safety-related SSC in 10 CFR 50.2, SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," and the prior application of reactivity control requirements. Notably, as written, the applicant's proposed PDC 26 would require an applicant referencing the TR to achieve stable, safe shutdown with margin following any accident using only safety-related systems, as well as having a minimum of two systems to achieve the four requirements set forth in the applicant's proposed PDC 26. These criteria, as written, obviate the need for a PDC like GDC 27. In considering the potential application of the applicant's proposed PDC 26 in a future submittal, the NRC staff may use the referenced justification in RG 1.232 to reach its finding. For the reasons documented here and detailed in RG 1.232, the NRC staff finds proposed PDC 26 to be acceptable.

The following set of PDCs in RG 1.232 relate to the reactor coolant boundary: 14, 30, 31, and 32. In general, the applicant's proposed changes included the use of the term "safety significant" instead of "important to safety," as outlined above. The applicant's proposed PDC 14 modifies the ARDC to state that it applies to the "…safety significant elements of the reactor coolant boundary." The applicant's proposed PDCs 30, 31, and 32 have similar language changes regarding the safety significance.

The applicant's proposed PDC 30 states that the quality standards used will be commensurate with the safety functions to be performed. The applicant's proposed PDC 31 adds the phrase "safety significance based on preventing brittle failure at the locations of the reactor coolant boundary which would have an impact on safety." The applicant's proposed PDC 32 also adds the phrase "safety significant for components."

The NRC staff reviewed the use of "safety significant" and "safety significant portions" in consideration of the KP-FHR design attributes discussed in Section 3.1.2 of this SE. As described in Section 3.2.3 of the TR, "safety significance" indicates SSCs that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. The applicant stated that the reactor coolant "system" is not necessary to ensure the health and safety of the public but rather that "components of the reactor coolant system" are necessary. Because of this, portions of the reactor coolant system may not be credited as a principal barrier to radioactive release. The applicant stated that portions of the reactor coolant boundary are credited to ensure that the fuel remains covered in a loss-of-coolant event. Therefore, the words "safety significant" are used to designate the portions of the reactor coolant boundary that are required to perform a coolant retention safety function. The NRC staff understands that this will have to be demonstrated by an applicant in future licensing documentation. As such, consistent with the condition discussed in Section 4.0 of this SE, the NRC staff finds the wording of the applicant's proposed PDCs 14, 30, 31, and 32, to be acceptable; provided the design features described in Section 1.1.2 of the TR are demonstrated to support the safety case of the KP-FHR design in a future licensing application. The applicant's proposed PDCs 14 and 30 also state that these safety significant portions of the reactor boundary will be subject to leakage detection, as necessary. Based on the design attributes of the KP-FHR design and the verification of these key design features supporting the safety case of the KP-FHR design in a future licensing application, the NRC staff finds this wording change to be acceptable.

The applicant's proposed PDC 32 also adds the word "monitoring" along with "inspection and functional testing of the reactor coolant boundary." The NRC staff finds this wording to be acceptable as monitoring may be another method of accomplishing the underlying purpose of PDC 32 to assess the reactor coolant boundary structural and leak-tight integrity.

The applicant's proposed PDC 33 makes two changes: one to add safety significant elements to the reactor coolant boundary language, and one to replace "fuel" with "system radionuclide release." The NRC staff's evaluation for these modifications is provided above with regards to PDCs 12, 17, and 26. Aside from these changes, the rationale for the applicant's proposed PDC 33 conforms with the rationale in RG 1.232 and therefore, the NRC staff finds the applicant's proposed PDC 33 to be acceptable.

The applicant's proposed PDC 34 and 35 are related. In both cases, the applicant has proposed to adopt the ARDC from RG 1.232 with modifications to accommodate design-specific provisions of the KP-FHR. PDC 34 replaces "fuel" with "system radionuclide release," as discussed above in relation to proposed PDCs 12, 17, 26, and 33, and further qualifies that only the "safety significant elements" (as described in more detail above related to proposed PDCs 14, 30, 31 and 32) of the reactor coolant boundary are subject to the proposed PDCs. The applicant stated in the bases that the KP-FHR design currently uses two decay heat removal systems, one safety-related and one not safety-related for normal operation, and AOOs. Denoting only the safety significant portions of the reactor coolant boundary is acceptable in this case as the primary safety function to be ensured by the system, residual heat removal, is captured by the other proposed change to the PDCs in that only the fuel and safety significant portions of the reactor coolant boundary are necessary to maintain system radionuclide release within limits. The design basis of any SSCs required to fulfill these PDCs would need to consider the full spectrum of design conditions and demonstrate that radionuclide

release limits were not exceeded using only safety significant elements of the reactor coolant system. The NRC staff believes the proposed wording is adequate but notes that the safety function associated with the system must be evaluated using only safety-related SSCs, in that they must be capable of meeting the acceptance criteria denoted by the PDCs, consistent with the rationale documented in RG 1.232.

This ties in with the functions assessed as part of PDC 35, in which the applicant has proposed two changes: use of "Passive residual heat removal" rather than "Emergency core cooling system" and "reactor internal structure" rather than "clad." The first change is semantic and not substantive, as the applicant stated there is no emergency core cooling system in the KP FHR but that the functions associated with that system – core cooling and removal of residual heat – are still performed. The second change has to do with the fuel form, which has no clad; use of the term "reactor internal structure" is sufficiently broad to capture the intended safety function associated with this PDC (to prevent damage to other SSCs that could inhibit core cooling). The applicant stated, "use of the word 'system' in this PDC includes those mechanical elements, flow paths, and features that support the function of residual heat removal." The NRC staff notes that this also includes a demonstration of adequate coolant inventory using only safety-related systems, in accordance with the rationale in RG 1.232. As discussed above, the changes to PDCs 34 and 35 do not change the essential safety functions discussed in the rationale of RG 1.232 ARDC 34 and 35, and the applicant's proposed PDCs are therefore, acceptable.

The NRC staff reviewed the applicant's proposed PDC 70 for the KP-FHR and the modifications from SFR-DC 71. The NRC staff notes that the PDC proposed by the applicant changes the wording in the SFR-DC so that the PDC refers to the reactor coolant instead of the primary coolant sodium. The NRC staff finds this change to be acceptable because the KP-FHR design does not use a sodium coolant and instead uses a molten salt coolant. Therefore, the PDC was changed to be applicable to this specific design.

The applicant's proposed PDC also removes cover gas from the requirements governing purity control. In the basis provided by the applicant for this PDC, it is noted that cover gas purity control is only one design option to maintain coolant purity, and that since there are no energetic interactions (i.e., sodium and cover gas contaminant moisture) present, cover gas purity does not need to be considered in the PDC. The NRC staff finds this to be acceptable because the overall purity of the primary salt coolant is still required to be maintained by the applicant's proposed PDC. Cover gas purity control may still be a means to demonstrate conformance to the applicant's proposed PDC, but alternative means may be used. Additionally, the NRC staff finds this change to the PDC to be acceptable because the KP-FHR design does not use sodium as its primary coolant and therefore does not need to consider the energetic sodium and cover gas contaminant moisture reaction.

Further, the NRC staff finds that the applicant's proposed PDC is necessary to ensure that the purity of the primary coolant of the KP-FHR design is maintained to limit and control chemical attack, fouling and plugging of passages, manage radionuclide concentrations, and control air or moisture ingress into the primary coolant. This is because the purity of the molten salt coolant may impact the structural integrity of the components within the reactor vessel and the flow paths that may impact decay heat removal. Salt purity may also impact radionuclide retention properties of the coolant. Because these are safety functions that rely on the purity of the coolant salt, the NRC staff finds the use of this PDC to be acceptable as it will provide reasonable assurance that the safety functions described above can be achieved, in part, by controlling the purity of the primary salt coolant.

Also, the NRC staff notes that part of the applicant's justification for this proposed PDC states that the initial coolant purity limits will consider the safety functions listed above.

The NRC staff finds this to be acceptable because these functions will be considered when developing the initial coolant purity limits which provides reasonable assurance that the purity control system will be able to maintain salt purity given the initial purity of the salt.

The NRC staff reviewed the applicant's proposed PDC 71 for the KP-FHR and the modifications from SFR-DC 72. The NRC staff finds it to be acceptable to replace "sodium" with "reactor coolant" in this PDC as the Kairos Power KP-FHR design does not use a sodium coolant. It instead uses a molten salt as the reactor coolant. The NRC staff finds this change to be acceptable because use of this PDC is appropriate for a molten salt coolant as it requires heating systems to remain in its liquid phase.

The applicant stated in the justification for proposed PDC 71, that heating systems will be provided for safety significant systems and components that contain, or could be required to contain, reactor coolant (i.e., primary coolant salt) in its liquid form. The applicant also stated that these heating systems and associated controls will ensure that the temperature distribution and rate of change of temperature are maintained within design limits assuming a single failure. The NRC staff finds this to be acceptable because the salt that the applicant has proposed to use as the reactor coolant is a solid at room temperature and requires heat in order to remain a liquid and ensure the reactor coolant remains within its design limits. The KP-FHR design relies on the salt to remove heat from the reactor core and in order to provide this function, the salt needs to be in its liquid phase. Therefore, the NRC staff finds the applicant's proposed PDC 71, to be acceptable because a PDC that requires maintaining the salt as a liquid (where safety significant) provides reasonable assurance that the reactor coolant can provide its function of removing heat from the reactor core. The NRC staff makes no finding on the adequacy of the heating systems, only that the applicant's proposed PDC is adequate to provide appropriate requirements for those systems.

The applicant's proposed PDC 71 also states that if plugging of a cover gas line due to condensate or plate out of reactor coolant aerosol or vapor could prevent accomplishing a safety function, the temperature control and associated corrective actions of that line shall be considered safety significant. The NRC staff finds this part of the proposed PDC to be acceptable because it provides reasonable assurance that the reactor coolant would not experience temperatures that could cause aerosols or vapors that may prevent the cover gas system from accomplishing a safety function. The NRC staff's evaluation for the modification from "important to safety" to "safety significant" is provided previously in this section.

The NRC staff reviewed the applicant's proposed PDC 73 for the KP-FHR and the modifications from SFR-DC 78. The applicant stated that it plans to use chemically compatible salts in its primary and intermediate heat transport systems. However, even though the applicant has proposed to use chemically compatible salts, the requirements of the proposed PDC that discuss the required passive barriers for incompatible salts, still apply. Use of chemically compatible salts would be a potential means to satisfy the requirements of the proposed PDC (i.e., if salts are demonstrated to be compatible then it is possible to demonstrate parts of the proposed PDC do not apply). Therefore, the NRC staff finds this section of the applicant's proposed PDC 73, to be acceptable.

The applicant's proposed PDC 73 also states that for chemically compatible salts a single passive barrier may be used given certain provisions. One of these provisions is that postulated leakage would not result in the failure of intended safety functions of SSCs that are safety significant. The NRC staff finds this to be acceptable because this provides reasonable assurance that in the case of leakage of a fluid into the primary coolant, it would not negatively impact intended safety functions of safety significant SSCs. The NRC staff's evaluation for the modification from "important to safety" to "safety significant" is provided earlier in this section.

Criteria in MHTGR-DC 70 and 71 in RG 1.232 relate to the reactor building design basis and inspection of the reactor building. In the applicant's proposed PDCs 74 and 75, which are based on MHTGR-DC 70 and 71 in RG 1.232, the applicant removed the text related to a pathway for release of reactor helium in the event of a depressurization event. The NRC staff finds the removal of the reference to helium depressurization events to be acceptable due to the KP-FHR using salt that operates at a low pressure. A reactor building will provide protection to the KP-FHR SSCs from external events to ensure passive heat removal. Therefore, the NRC staff finds the inclusion of both proposed PDC 74 and 75, to be acceptable.

The applicant has stated that SFR-DC 70 is not applicable to the KP-FHR design. Guidance in RG 1.232, Appendix B, SFR-DC 70 states that the purpose of this design criteria is to ensure design conditions of the intermediate coolant boundary are not exceeded during normal operations and AOOs, and the integrity of the primary coolant boundary is maintained during postulated accidents. This design criteria along with SFR-DC-75 and SFR-DC-76, describe the three design functions of the intermediate loop in a SFR: (1) ensures that the intermediate system doesn't impact the safety of the primary system; (2) ensures radioactivity from the primary system doesn't transfer to the power conversion system; and (3) ensures the design of the intermediate system minimizes the possibility of a large, uncontrolled release of sodium. The SFR-DC-77 provides supplementary design criteria to ensure that the intermediate system can perform the three design functions throughout the lifetime of the plant.

The applicant stated that SFR-DC-70, SFR-DC-75, SFR-DC-76, and SFR-DC-77 are not necessary for the KP-FHR because the design of the intermediate loop inherently meets the three design functions of the SFR design criteria. The applicant stated that the intermediate system does not impact the safety of the primary system. The existence of an intermediate loop ensures that there is defense-in-depth to prevent radioactive material from being introduced into the power conversion system. Finally, the intermediate loop of the KP-FHR utilizes molten salt and as such there is no possibility for a large release of sodium. The NRC staff agrees that SFR-DC 70, SFR-DC-75, SFR-DC-76, and SFR-DC-77 are not necessary for the KP-FHR as long as the assumptions for the intermediate loop are reflected in the licensing documents. These assumptions are described in Section 4.0 of this SE.

The applicant has stated that SFR-DC 73 is not applicable to the KP-FHR design. Guidance in RG 1.232, Appendix B, provides the rationale for this criterion which is to preclude the adverse chemical reactions between sodium and air, and sodium and concrete. Additionally, the rationale states that an additional design criterion is suggested because the GDC does not contain a similar criterion to account for the high chemical activity of sodium with common plant materials such as water, air, and concrete. However, the KP-FHR does not use sodium as a coolant and the proposed salts do not have high chemical activity with materials such as water, air, or concrete. Therefore, the NRC staff finds it to be acceptable to not apply SFR-DC 73 to the KP-FHR.

The applicant has stated that SFR-DC 74 is not applicable to the KP-FHR design. Guidance in RG 1.232, Appendix B, provides the rationale for this criterion, which is to preclude the adverse chemical reaction between sodium and water coolants. Because the KP-FHR does not use sodium as a coolant, the NRC staff finds it to be acceptable to not apply SFR-DC 74 to the KP-FHR.

The applicant has stated that SFR-DC 79 is not applicable to the KP-FHR design. The applicant stated that because of the different reactor coolant chemistry, certain energetic interactions with the coolant are not thermodynamically favored in the KP-FHR. This means that cover gas inventory maintenance is not a safety function. Guidance in RG 1.232, Appendix B, states that the cover gas performs an important to safety function by protecting a sodium coolant from chemical reactions.

As stated in the bases for KP-FHR PDC 70, cover gas purity may be a design solution to meet chemistry control requirements of the reactor coolant and demonstrate compliance with PDC 70. However, since there may be other means to meet reactor coolant purity requirements, it is not specifically required in the applicant's proposed PDC 70. In SFR-DC 79 it states that the maintenance of cover gas inventory is needed to ensure primary coolant sodium design limits are not exceeded. Guidance in NUREG-1368 (Reference 7), Section 7.3.6.3, "Impurity-Monitoring System," states that primary system purity design limits "…should be based on consideration of chemical attack, fouling and plugging of passages, radioisotope concentrations, and detection of sodium-water interactions."

The NRC staff finds it to be acceptable to not apply SFR-DC 79 to the KP-FHR for two reasons: (1) the proposed reactor coolant for the KP-FHR does not have a high chemical activity like sodium; and (2) the applicant has stated that cover gas purity control is not required to maintain reactor coolant purity and therefore considerations such as chemical attack, fouling, radionuclide concentrations, and air/moisture ingress will be addressed via the reactor coolant purification system and the applicant's proposed PDC 70.

3.3.4 Conclusion

The applicant developed a set of PDCs based on the GDCs and guidance in RG 1.232 and stated they "reflect the key design features of the KP-FHR technology and provide an appropriate set of requirements to facilitate the design and licensing of the KP-FHR." The applicant requested that the NRC staff approve the proposed PDCs for use by future licensing applicants under 10 CFR Parts 50 or 52 given the details of the KP-FHR remain consistent with the key design attributes identified in the TR.

3.3.5 NRC Staff Evaluation

The NRC staff agrees that the applicant has considered each of the design aspects presented in RG 1.232 and has developed PDCs based on the guidance presented in the RG, as discussed in Section 3.3.3.1 of this SE. The proposed PDCs are sufficiently broad to provide an appropriate set of requirements for the KP-FHR, subject to the limitations and conditions below.

4.0 LIMITATIONS AND CONDITIONS

The NRC staff imposes the following limitations and conditions regarding the TR:

1. (**Section 3.1.2**) As presented in the TR, there are key design features without which the proposed PDC would not be applicable or encompass the full set of necessary design criteria. Therefore, a KP-FHR design referencing the TR must have the following:

    - A "chemically stable molten fluoride salt mixture" coolant.

    - TRISO fuel particles and fuel pebbles that, combined with other design features as applicable, demonstrate functional containment performance criteria consistent with SECY-18-0096 and applicable regulatory dose requirements.

    - An intermediate coolant loop using a coolant that is compatible with reactor coolant, and that is demonstrated not to have a safety significant impact on the primary system.

    - "Near-atmospheric" primary coolant pressures.

- The ability to ensure core cooling by maintaining coverage of the fuel within the reactor core with coolant.

If other key design features are identified by the applicant that could necessitate additional PDCs, those PDCs would be subject to the NRC staff's review, independent of the TR.

2. (**Section 3.1.2**) The proposed scope of a manufacturing license that would reference the TR and how the proposed PDC would be applicable is not sufficiently clear. As such, any use of the TR in a ML application would be conditional on a related license application with a clear scope (a CP, COL, or DC application).

3. (**Section 3.3.2**) The development of this SE was informed by guidance in DG-1353 and NEI 18-04. However, use of this guidance is not yet approved by the NRC. Further, the methodology described in NEI 18-04 is an integral process that requires a full understanding of all plant SSCs and their role in the probabilistic risk assessment and would need to appropriately consider all aspects of plant safety. Therefore, use of the TR by an applicant is conditional on the NRC's approval of the guidance in DG-1353 and NEI 18-04, with or without modification, and staff's further approval of any deviations from the approved guidance requested by the applicant.

4. (**Section 3.3.3**) Use of the term "safety-related" as described in the TR is narrowly applicable to the context discussed herein and must include SSCs designated as safety-related as defined in NEI 18-04 and endorsed by the NRC in DG-1353 as well as any SSCs that meet the definition of 10 CFR 50.2 as applicable to the broader future application referencing the TR.

5.0 CONCLUSION

Based on the above evaluation, the NRC staff concludes that Kairos Power has provided a sufficient set of PDCs for establishing requirements for the KP-FHR design, subject to the limitations and conditions listed in Section 4.0 of this SE. The proposed PDCs meet the underlying purpose and technical rationale of the ARDC in RG 1.232. Conformance with these PDCs, subject to the limitations and conditions of the TR, establish the necessary design, fabrication, construction, testing, and performance requirements for safety significant SSCs to provide reasonable assurance that a KP-FHR could be operated without undue risk to the health and safety of the public.

Principal Contributor: Boyce Travis, NRR

Date: May 22, 2020.

6.0 REFERENCES

1. U.S. Nuclear Regulatory Commission, "Guidance for Developing Principal Design Criteria for Non-Light Water Reactors," RG 1.232, Revision 0

2. Nuclear Energy Institute, "Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development," NEI 18-04 (Draft)

3. U.S. Nuclear Regulatory Commission, "Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors," DG-1353, April 2019

4. U.S. Nuclear Regulatory Commission, "Functional Containment Performance Criteria for Non Light-Water-Reactors," SECY-18-0096, October 16, 2018

5. U.S. Nuclear Regulatory Commission, "Staff Requirements- SECY-18-0096 Functional Containment Performance Criteria for Non-Light-Water-Reactors," December 4, 2018

6. U.S. Nuclear Regulatory Commission, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," SECY-94-084, March 28, 1994

7. U.S. Nuclear Regulatory Commission, Preapplication Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Liquid-Metal Reactor, NUREG-1368, February 1994

Section B

KP-TR-003-NP-A

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Principal Design Criteria for the Kairos Power Fluoride Salt-Cooled High Temperature Reactor

Topical Report

Revision 1
July 2019

Non-Proprietary

REVISION LOG

Rev	Description of Change	Date
0	Initial Issuance	December 2018
1	Revised to address NRC questions, provided by email dated May 23, 2019, on the initial revision.	July 2019

Executive Summary

This topical report summarizes the methodology for development of the principal design criteria for the Kairos Power Fluoride Salt-Cooled, High Temperature Reactor. The principal design criteria are developed based on the key design features of the KP-FHR technology and used the guidance of Regulatory Guide 1.232, “Guidance for Developing Principal Design Criteria for Advanced (Non-Light Water) Reactors.” The resultant design criteria are comprehensive, reflect the key design features of the technology, and provide future license applicants under 10 CFR 50 or 10 CFR 52 a basis for design of the KP-FHR. Kairos Power is requesting Nuclear Regulatory Commission review and approval of these design criteria for use by future applicants.

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ABBREVIATIONS

Abbreviation or Acronym	Definition
AOO	Anticipated Operational Occurrences
ARDC	Advanced Reactor Design Criteria
CFR	Code of Federal Regulations
DC	Design Criteria
DOE	Department of Energy
EAB	Exclusion Area Boundary
ECCS	Emergency Core Cooling System
FHR	Fluoride Salt-Cooled High-Temperature Reactor
GDC	General Design Criteria
KP-FHR	Kairos Power Fluoride Salt-Cooled High Temperature Reactor
LWR	Light Water Reactor
MHTGR	Modular High Temperature Gas-Cooled Reactor
MHTGR-DC	MHTGR Design Criteria
NRC	Nuclear Regulatory Commission
PDC	Principal Design Criteria
RG	Regulatory Guide
SARRDL	Specified Acceptable System Radionuclide Release Design Limit
SAFDL	Specified Acceptable Fuel Design Limit
SFR	Sodium-Cooled Fast Reactor
SFR-DC	SFR Design Criteria
SSC	Structures, Systems, and Components
TRISO	Tri-structural Isotropic

1 INTRODUCTION

Kairos Power LLC (Kairos Power) is pursuing the design, licensing, and deployment of the Kairos Power Fluoride Salt-Cooled, High Temperature Reactor (KP-FHR). To support these objectives, Kairos Power is developing Principal Design Criteria (PDC) applicable to the KP-FHR design. Nuclear Regulatory Commission (NRC) regulations in 10 CFR 50.34(a)(3)(i) require that applicants for a construction permit include the PDC for a facility. Similarly, NRC regulations in 10 CFR 52.47(a)(3)(i), 10 CFR 52.79(a)(4)(i), 10 CFR 52.137(a)(3)(i), and 10 CFR 52.157(a) require that applications for standard design certifications, combined licenses, standard design approvals, and manufacturing licenses include the PDC for a facility.

NRC regulations in 10 CFR 50, Appendix A provide General Design Criteria (GDC) that establish the minimum requirements for PDC for light water reactors (LWRs). 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. NRC has published Regulatory Guide (RG) 1.232, “Guidance for Developing Principal Design Criteria for Non-Light Water Reactors” (Reference 1), which provides guidance for establishing the PDC for non-light water reactor designs. RG 1.232 also includes PDC for two specific non-LWR designs, the Sodium-Cooled Fast Reactor (SFR) and the Modular High-Temperature Gas-Cooled Reactor (MHTGR).

This report provides the PDC for the KP-FHR which are developed using the guidance in RG 1.232 and in consideration of unique KP-FHR design attributes. Kairos Power requests NRC review and approval of these PDC to be used by applicants of the KP-FHR design for standard design certifications, combined licenses, standard design approvals, and manufacturing licenses under the applicable regulations in 10 CFR 52; or limited work authorizations, construction permits and operating licenses under 10 CFR 50. The approval of these PDC are based on the existence of the key design features of the KP-FHR technology identified in Section 1.1.2. The demonstration that the KP-FHR design satisfies these PDC will be provided within the license application documents (e.g., safety analysis reports) required to be submitted by the cited regulations.

1.1 DESIGN FEATURES

1.1.1 DESIGN BACKGROUND

To facilitate NRC review and approval of this report for use by future applicants, key design features are provided in Section 1.1.2 which are inherent to the KP-FHR technology. These features are not expected to change during the design development by Kairos Power and provide the basis to support the safety review of this report. Should fundamental changes occur to these key design features, or new or revised regulations be promulgated that affect the PDC for the KP-FHR, such changes would be reconciled and addressed in license application submittals.

The KP-FHR is a U.S.-developed Generation IV advanced reactor technology. In the last decade, U.S. national laboratories and universities have developed pre-conceptual Fluoride High-Temperature Reactor (FHR) designs with different fuel geometries, core configurations, heat transport system configurations, power cycles, and power levels. More recently, University of California at Berkeley developed the Mark 1 pebble-bed FHR, incorporating lessons learned from the previous decade of FHR pre-conceptual designs (Reference 2). Kairos Power has built on the foundation laid by Department of Energy (DOE)-sponsored university Integrated Research Projects (IRPs) to develop the KP-FHR.

Table 1 compares and contrasts the major design attributes of the KP-FHR with those reflected in RG 1.232, i.e. the SFR and the MHTGR. The purpose of this table is to aid in the understanding of the KP-FHR PDC development and the rationale for changes to wording from the RG. It should be emphasized that the KP-FHR contains design features similar in nature to those found in the SFR or MHTGR, and, with a notable exception, does not add new or unique features from those present in the SFR or MHTGR designs. The exception from these designs is the use of pebble based fuel operating in an essentially continuous refueling process during normal operations.

Although not intended to support the findings necessary to approve the PDC, additional design description information is provided in the technical report "Design Overview of the Kairos Power Fluoride Salt-Cooled, High Temperature Reactor" (Reference 3).

1.1.2 KEY DESIGN FEATURES OF THE KP-FHR

The KP-FHR is a high temperature reactor with molten fluoride salt coolant operating at near-atmospheric pressure. The fuel in the KP-FHR is based on the tri-structural isotropic (TRISO) high-temperature, carbonaceous-matrix coated particle fuel (originally developed for high-temperature gas-cooled reactors) in a pebble fuel element. Coatings on the particle fuel provide retention of fission products. The reactor coolant is a chemically stable molten fluoride salt mixture, $2LiF:BeF_2$ (Flibe with enrichment of the ^7Li isotope) which also provides retention of fission products that escape from any fuel defects. The KP-FHR includes a fuel handling system which extracts fuel from the active core and, based on inspection, either inserts it back into the active core or directs the fuel to spent fuel storage. While outside the active core, the fuel is not critical and producing only decay heat and is not covered by salt. The functional containment is still met by the layers of the TRISO fuel particle.

A primary coolant loop circulates the reactor coolant using pumps and transfers the heat to an intermediate coolant loop via a heat exchanger. The pumped flow intermediate coolant loop utilizes a nitrate salt, chemically compatible with reactor coolant, and transfers heat from the reactor coolant to the power conversion system through a steam generator. The design includes two decay heat removal systems. A normal decay heat removal system is used following normal shutdowns and anticipated operational occurrences. A separate passive decay heat removal system, which along with natural circulation in the reactor vessel, removes decay heat in response to a design basis accident and does not rely on electrical power.

The KP-FHR design uses a functional containment approach similar to the MHTGR instead of the typical LWR low-leakage, pressure retaining containment structure. The KP-FHR functional containment design objective is to meet 10 CFR 50.34 (10 CFR 52.79) offsite dose requirements at the plant's exclusion area boundary (EAB) with margin. A functional containment is defined in RG 1.232 as a "barrier, or set of barriers taken together, that effectively limit the physical transport and release of radionuclides to the environment across a full range of normal operating conditions, AOOs, and accident conditions." RG 1.232 includes an example design criterion for the functional containment (MHTGR Criterion 16). As also stated in RG 1.232, the NRC has reviewed the functional containment concept and found it "generally acceptable," provided that "appropriate performance requirements and criteria" are developed. The NRC staff has developed a proposed methodology for establishing functional containment performance criteria for non-LWRs, which is presented in SECY-18-0096. This SECY document has been approved by the Commission.

The functional containment approach for the KP-FHR is to control radionuclides primarily at their source within the coated fuel particle under normal operations and accident conditions without requiring active design features or operator actions. The KP-FHR design relies primarily on the multiple barriers within the TRISO fuel particles and fuel pebble to ensure that the dose at the site boundary as a consequence of postulated accidents meets regulatory limits. However, in contrast to the MHTGR, the KP-FHR molten salt coolant also serves as a distinct barrier providing retention of fission products that escape the fuel particle and fuel pebble barriers. This additional retention is a key feature of the enhanced safety and reduced source term in the KP-FHR.

1.2 REGULATORY REVIEW

As previously noted, facilities licensed under 10 CFR Part 50, including both LWRs and non-LWRs, are required to describe the PDC in their preliminary safety analysis report supporting a construction permit application as described in 10 CFR 50.34(a)(3)(i). Likewise, applicants for standard design certifications, combined licenses, standard design approvals, and manufacturing licenses must include the PDC for a facility as described in 10 CFR 52.47(a)(3)(i), 10 CFR 52.79(a)(4)(i), 10 CFR 52.137(a)(3)(i), and 10 CFR 52.157(a).

The GDC in 10 CFR 50, Appendix A, provide minimum requirements for a plant's PDC, which establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components (SSCs) that are safety significant. The GDC in Appendix A have served as a key part of the regulatory framework for LWRs for many years. As noted above, 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. With the advent of new non-light water reactor designs, the NRC and U.S. Department of Energy (DOE) implemented a joint initiative to assess the GDC and determine the extent to which they apply to non-LWR designs and to propose amended and/or additional design criteria that address non-LWR design features. The results of this effort culminated in NRC RG 1.232.

RG 1.232 provides a set of advanced reactor design criteria (ARDC), which serve the same purpose for non-LWRs as the GDC serve for LWRs. The non-LWR designs considered during the joint initiative leading to the development of this RG included SFRs, lead-cooled fast reactors, gas-cooled fast reactors, MHTGRs, fluoride high-temperature reactors, and molten salt reactors. The ARDC are intended to be technology inclusive to align with the six technologies above. In addition to the technology-inclusive ARDC, RG 1.232 provides two sets of technology-specific, non-LWR design criteria. The two technology-specific design criteria are provided for the SFR and the MHTGR. The PDC provided for the SFR and MHTGR designs are referred to as the SFR design criteria (SFR-DC) and the MHTGR design criteria (MHTGR-DC), respectively. As indicated in RG 1.232, the NRC intends that the ARDC apply to the six advanced reactor technology types identified in the DOE report; however, in some instances, one or more of the criteria from the SFR-DC or MHTGR-DC may be more applicable to a design or technology than the ARDC.

Relevant excerpts from the guidance for development of PDC from RG 1.232 are provided below:

- "Since the GDC in 10 CFR 50 Appendix A are not regulatory requirements for non-LWR designs but provide guidance in establishing the PDC for non-LWR designs, non-LWR applicants would not need to request an exemption from the GDC in 10 CFR Part 50 when proposing PDC for a specific design."
- "Applicants may use this RG to develop all or part of the PDC and are free to choose among the ARDC, SFR-DC, or MHTGR-DC to develop each PDC after considering the underlying safety

basis for the criterion and evaluating the rationale for the adaptation described in this RG. For example, Fluoride High Temperature Reactors (FHRs) are molten salt reactors that use TRISO fuel, which is the same fuel used for MHTGR technologies. An FHR designer could use the MHTGR-DC where appropriate for the design. Another example is the MSRs that use liquid fuel. An MSR designer may need to develop new PDC for liquid fuel and systems to support this design."

- "In each case, it is the responsibility of the designer or applicant to provide not only the PDC for the design but also supporting information that justifies to the NRC how the design meets the PDC submitted, and how the PDC demonstrate adequate assurance of safety."
- "Finally, the non-LWR design criteria as developed by the NRC staff are intended to provide stakeholders with insights into the staff's views on how the GDC could be interpreted to address non-LWR design features; however, these are not considered to be final or binding on what may eventually be required from a non-LWR applicant."

The PDC provided in this report are intended to be used by future license applicants using the KP-FHR design to satisfy the aforementioned regulatory requirements.

2 PDC DEVELOPMENT METHODOLOGY

This section describes the process used by Kairos Power to develop the PDC for the KP-FHR. The starting point for this process is a review of the ARDC from RG 1.232, Appendix A for its relevance to the key design features of the KP-FHR technology. Each ARDC in Appendix A of RG 1.232 is reviewed for applicability to the KP-FHR design, considering the underlying safety basis for the ARDC and the supporting information in Appendix A of RG 1.232. Note that in some cases, the ARDC in RG 1.232 adopts the GDC from 10 CFR 50, Appendix A without change.

Where the Kairos Power review of the ARDC concluded that the ARDC could be directly adopted for the KP-FHR, then the ARDC is selected as the PDC for the KP-FHR. Because of similarities with certain features of SFR and MHTGR technologies, the SFR-DC and MHTGR-DC are also reviewed for relevancy to the KP-FHR and to determine whether changes from the ARDC identified in RG 1.232, Appendix B or Appendix C for these technologies should be considered for inclusion in the PDC for the KP-FHR.

For those ARDC that did not fully apply to the key design features of the KP-FHR, then the SFR-DC and MHTGR-DC are assessed to determine if either could be directly adopted. If either the SFR-DC or MHTGR-DC are representative of the KP-FHR technology, then the one that is most representative is selected as the PDC.

If none of the ARDC, the SFR-DC, or the MHTGR-DC are adopted as written, a judgment is made as to which of the three is most representative of the KP-FHR key design features. This assessment is based on technical relevance and the amount of modification that would be necessary to conform the criteria to be representative of the KP-FHR. Modifications are then made to reflect the design of the KP-FHR and the departures from the underlying criteria are annotated.

In a number of cases, the ARDC addressed a system or component that does not exist nor would be necessary or desirable to implement the KP-FHR technology. Examples include a containment building or emergency core cooling system. In these instances, these ARDC are determined to be not applicable for the KP-FHR PDC, similar to the conclusion for the MHTGR-DC in RG 1.232.

RG 1.232 includes a number of SFR and MHTGR technology-specific additional design criteria (numbered 70 and higher) in Appendix B and Appendix C. Each of these technology specific criteria are evaluated for applicability to the KP-FHR. As previously noted, the KP-FHR technology includes some features similar to those of the SFR and MHTGR.

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

The RG 1.232 review method described above is performed by Kairos Power personnel knowledgeable in the KP-FHR technology and development of the ARDC. The results of the review are documented along with a basis for selection of the KP-FHR PDC. The results of the review and KP-FHR PDC are provided in Section 3. The KP-FHR PDC are internally reviewed by Kairos Power engineering and licensing personnel. Additionally, an external team of industry experts knowledgeable of the KP-FHR, SFR, MHTGR, and LWR technologies, performed an independent review of the PDC. Feedback from this review is used to inform the results. Figure 1 provides a flow chart of the methodology described above to develop the PDC.

3 RESULTS

3.1 SUMMARY OF RESULTS

This section provides a summary of the results of the evaluation of the criteria in RG 1.232 and their applicability to the KP-FHR. Further details are provided in later sections and in the Appendices. The results are summarized in the following sections consistent with RG 1.232:

Section I—Overall Requirements (Criteria 1–5)

Section II—Multiple Barriers (Criteria 10–19)

Section III—Reactivity Control (Criteria 20–29)

Section IV—Fluid Systems (Criteria 30–46) for ARDCs, and SFR-DC, entitled Heat Transport Systems (Criteria 30-46) for MHTGR-DC

Section V—Reactor Containment (Criteria 50–57)

Section VI—Fuel and Radioactivity Control (Criteria 60–64)

Section VII—Additional SFR-DC (Criteria 70–77)

Section VII—Additional MHTGR-DC (Criteria 70–72)

Section I—Overall Requirements (Criteria 1–5)

These design criteria involve quality standards and records, design bases for protection against natural phenomena, fire protection, environmental and dynamic effects, and sharing of structures, systems, and components. There are five requirements in this category and they are general non-design specific requirements. In all five cases the ARDC are adopted with minor modifications.

Section II—Multiple Barriers (Criteria 10–19)

These criteria involve the barriers to the release of radioactivity, specifically reactor design, reactor inherent protection, suppression of reactor power oscillations, instrumentation and control, reactor coolant boundary, reactor coolant system design, containment design, electric power systems, inspection and testing of electric power systems, and control room. These criteria are more specific to the key design features of the technology. As a result, some modifications to the design criteria in this category are necessary to align with the KP-FHR technology.

There are ten requirements in this category. In one case the ARDC is directly applicable with no modification, in six other cases the ARDC are applicable with modifications, and in three cases the MHTGR-DC are applicable with modifications.

Section III—Reactivity Control (Criteria 20–29)

These criteria involve protection system functions, protection system reliability and testability, protection system independence, protection system failure modes, separation of protection and control systems, protection system requirements for reactivity control malfunctions, reactivity control systems, combined reactivity control systems capability, reactivity limits, and protection against anticipated operational occurrences.

There are nine requirements in this category. These design criteria are not design specific. In five cases, the ARDC are directly applicable and adopted without modification, in two cases the ARDC are

adopted with modification, in one case the MHTGR-DC is applicable with no modification, and in one case the MHTGR-DC is relevant and adopted with a modification.

Section IV—Fluid Systems (Criteria 30–46) for ARDCs, and SFR-DC, entitled Heat Transport Systems (Criteria 30-46) for MHTGR-DC

These design criteria relate to fluid systems used for advanced reactors and sodium fast reactors. The criteria in this section are called heat transport systems for the MHTGR because it uses a gas for cooling. These criteria relate to the quality of reactor coolant boundary, fracture prevention of reactor coolant boundary, inspection of reactor coolant boundary, reactor coolant inventory maintenance, residual heat removal, emergency core cooling, inspection of emergency core cooling system, testing of emergency core cooling system, containment heat removal, inspection of containment heat removal system, testing of containment heat removal system, containment atmospheric cleanup, inspection of containment atmosphere cleanup, testing of containment atmosphere cleanup systems, structural and equipment cooling, inspection of structural and equipment cooling systems, and testing of structural and equipment cooling systems.

There are seventeen design criteria in this category. In two cases the ARDC are relevant and adopted with no changes, in six cases the ARDC are relevant but require minor modifications, in two cases the MHTGR are relevant and adopted with no changes, in one case the MHTGR-DC is relevant and adopted with a minor modification, and in six cases (Criteria 38-43) the criteria are considered to be not applicable (similar to the conclusions for the MHTGR-DC in RG 1.232). These latter criteria are not necessary due to specific reference to containment structures, which are not part of the KP-FHR design due to reliance on the functional containment approach (Section 1.1.2).

Section V—Reactor Containment (Criteria 50–57)

These design criteria relate to reactor containment structures. These criteria 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. All eight of these criteria are deemed to be not applicable, which is consistent with the MHTGR-DC.

Section VI—Fuel and Radioactivity Control (Criteria 60–64)

These design criteria relate to fuel and radioactivity control, including control of releases of radioactive material to the environment, fuel storage and handling and radioactivity control, prevention of criticality in fuel storage and handling, monitoring fuel and waste storage, and monitoring radioactivity releases. There are five criteria. In three instances, the ARDC are directly relevant and adopted as is, in one case the ARDC is adopted with modifications, and in one case the MHTGR-DC is adopted with no modifications.

Section VII—Additional SFR-DC (Criteria 70–79)

These design criteria are technology specific to SFRs and relate to the intermediate coolant system, primary coolant and cover gas purity control, sodium heating systems, sodium leakage detection and reaction prevention and mitigation, sodium/water reaction prevention/mitigation, quality of the intermediate coolant boundary, fracture prevention of the intermediate coolant boundary, inspection of the intermediate coolant boundary, primary coolant system interfaces, and cover gas inventory maintenance.

Of the ten technology-specific SFR-DC, three criteria are applicable and require modification to be applicable to the KP-FHR. These requirements are primary coolant and cover gas purity, sodium heating system, and primary coolant system interfaces.

Section VII—Additional MHTGR-DC (Criteria 70–72)

These design criteria are technology specific to MHTGRs and relate to the reactor vessel and reactor system structural design basis, reactor building design basis, and provisions for periodic reactor building inspection. There are three design criteria, one of which applies to the KP-FHR without modifications, and two of which apply to the KP-FHR with modifications.

3.2 SUMMARY OF CHANGES TO THE ARDC

An overall summary of the results of the PDC development is provided in Table 2. As can be seen from the table, many of the criteria from the ARDC apply. Of the ones that do not apply, in most cases the analogous MHTGR-DC are a closer fit than the SFR-DC. This is expected in that the key design features of the KP-FHR are generally closer to the design features of an MHTGR than to an SFR.

As discussed in Section 2, a methodical review was performed to be sure that the PDC collectively provide a comprehensive design and regulatory framework for the KP-FHR. This review concluded that all the major features of the KP-FHR were addressed by a PDC. Of particular note is the pebble-based design with continuous refueling (Section 1.1.1), which is considered to be a unique design feature of the KP-FHR. This feature is sufficiently addressed by PDC 61 (Fuel storage and handling and radioactivity control) and PDC 62 (Prevention of criticality in fuel storage and handling), which address cooling and criticality of the spent fuel, respectively. This review confirmed that there are no design attributes in the KP-FHR that are not addressed by the selected set of PDC. Accordingly, no new PDC were identified as necessary.

Table 3 provides a summary of the primary modifications made to the ARDC, SFR-DC, and MHTGR-DC and to which KP-FHR PDC they apply.

There are several common themes in the modifications to the ARDC, SFR-DC or MHTGR-DC that are made for KP-FHR. These are described in the subsections below.

The following terms are uniquely defined below for the KP-FHR and are provided to aid in the understanding of the development of the KP-FHR PDC.

Reactor Core – the region within the reactor vessel where the fuel pebbles are critical during operation.

Reactor System – is the reactor core and associated heat removal, control, and protection systems.

Reactor Coolant Boundary – all those reactor components connected to and including the reactor vessel and the first physical interface or isolable geometry which separates the reactor coolant and cover gas from other systems.

3.2.1 SPECIFIED ACCEPTABLE SYSTEM RADIONUCLIDE RELEASE DESIGN LIMIT

In the KP-FHR PDC, the phrase "Specified Acceptable Fuel Design Limits (SAFDLs)" has been replaced with the term from the MHTGR-DC in RG 1.232; "Specified Acceptable System Radionuclide Release Design Limit (SARRDL)."

SAFDLs, as used in the General Design Criteria (GDC) and Advanced Reactor Design Criteria (ARDC) are deterministic design limits which prevent additional fuel failures during anticipated operational occurrences (AOOs). Due to the nature of TRISO fuel, it is not possible (and unnecessary from an offsite dose perspective) to ensure zero incremental fuel failures during an AOO.

SARRDLs were developed for the MHTGR to address the need for a performance limit that reflects the performance and characteristics of coated particle fuel. In the analysis of potential offsite dose consequences for certain MHTGR postulated accidents, two limits were identified that significantly influenced the offsite dose: (1) the radionuclide concentration in circulating coolant and (2) radionuclides condensed on surfaces of the reactor helium pressure boundary. This is a result of the high pressure of the MHTGR coolant which drives much or all of this activity out of the reactor coolant (helium) pressure boundary and the reactor building during postulated rapid depressurization events. These limits are traditionally interpreted to be the SARRDLs for the MHTGR. Due to the different design, these limits do not apply (exactly) to the KP-FHR, as discussed below.

SARRDLs for the KP-FHR will fulfill the same intent described in RG 1.232 and will be established so that while some small increase in circulating radionuclide inventory may occur during an AOO, (1) the consequences of the most limiting design basis accident does not exceed the siting regulatory dose limits criteria at the exclusion area boundary (EAB) and low-population zone (LPZ), and (2) the 10 CFR 20.1301 annualized dose limits to the public are not exceeded at the EAB for normal operation and AOOs. In addition, the KP-FHR coolant is effective at retaining fission products and results in the normal operation coolant radioactivity level increasing slowly over time. Therefore, a normal operation radioactivity limit will be established that includes consideration of the anticipated small increases in radioactivity associated with AOOs. The normal operation radioactivity limit establishes an initial condition which provides assurance that the dose limits described above are met.

The other important difference in establishing SARRDLs for the KP-FHR originates from the low-pressure, single phase reactor coolant operation and the ability of the reactor coolant to retain selected fission products released from failed TRISO fuel particle layers. This indicates that the consequence of reactor coolant boundary leakage will depend on the location of the leakage rather than the entire connected set of equipment that may contain coolant. This may also result in establishment of different limits on circulating activity in the coolant and circulating activity in the cover gas.

3.2.2 FUNCTIONAL CONTAINMENT

As discussed in Section 1.1.2, the KP-FHR is adopting a functional containment and does not include a pressure-retaining containment structure as part of its design. The ARDC that pertain to a pressure-retaining containment structure are not adopted or are modified to reflect the KP-FHR functional containment.

3.2.3 SAFETY SIGNIFICANCE

The NRC staff defines “important to safety” generally in 10 CFR 50 Appendix A as “structures, systems, and components (SSCs) that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.” RG 1.232 provides guidance on development of PDC in lieu of Appendix A GDC. In an effort to clarify the scope of SSCs that perform functions that are safety related or non-safety-related but warranting special treatment, including those that perform functions that are necessary to meet defense-in-depth criteria, draft NEI-18-04

Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development) (Reference 5) explicitly defines the term "safety significant." The use of this term is endorsed in NRC's draft regulatory guide, DG-1353, and its use is judged to more consistently describe the SSCs that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. Accordingly, "safety significant" is used in defining KP-FHR PDC in lieu of "important to safety" as used in Appendix A for GDC.

3.2.4 SAFE SHUTDOWN

Modifications are made to several of the ARDC to replace the phrase "an orderly shutdown and cooldown" with "the ability to achieve and maintain safe shutdown". This change is made to remove the implied temperature of "cooldown." The KP-FHR is a high-temperature reactor, with the reactor coolant melting temperature significantly above ambient temperatures. RG 1.232 Appendix A Criterion 26 rationale and SECY-94-084 (Reference 6) describe a "safe shutdown" condition as one where the reactor is subcritical, there is sufficient decay heat removal, and radioactive materials are contained. This modification is made in a number of places in the KP-FHR PDC.

3.2.5 REMOVAL OF REFERENCE TO NON KP-FHR TERMS

In cases where the MHTGR-DC are adopted, the reference to helium coolant is removed because the KP-FHR does not use helium coolant and instead uses molten salt (Flibe). In several cases where an SFR-DC is adopted, reference to sodium coolant is removed because the KP-FHR does not use sodium as a coolant.

3.2.6 REACTOR COOLANT BOUNDARY

Modifications are made to nine PDC related to the reactor coolant boundary. These changes modify the requirements to apply only to the safety-significant portions of the reactor coolant boundary. The reason for this modification is based on the consequence of leakage as related to the location of leaks or breaks within the components containing reactor coolant. The KP-FHR utilizes a low-pressure, inert reactor coolant as well as a low-pressure, inert cover gas. Certain components on the reactor coolant boundary (for example those below a level which could expose the fuel) have a greater significance than other components in the event of an unmitigated coolant leak, and the quality standards should reflect this fact.

3.3 DETAILED KP-FHR PDC RESULTS

The detailed final results of the development of the KP-FHR PDC are provided in Appendix A and Appendix B of this report. Appendix A provides the PDC for the KP-FHR. Appendix B identifies those RG 1.232 PDC which are determined to be not applicable to the KP-FHR design. The PDC numbering system used herein is consistent with 10 CFR 50 Appendix A, which is also used in the appendices to RG 1.232. An exception to the numbering exists with respect to the technology specific PDC for SFR and MHTGR, which are re-numbered starting with PDC 70 as they are in the appendices to RG 1.232.

The detailed evaluation results are organized in a tabular form for each PDC as follows:

Title: This content reflects the number and the title of the PDC. In most cases, the titles from the RG 1.232 ARDC apply. In some cases, the title is changed to reflect KP-FHR design features.

KP-FHR PDC: This content reflects the final PDC wording for the KP-FHR.

Position: This content provides a determination of whether a given ARDC, MHTGR-DC, or SFR-DC is relevant and whether it is adopted with or without changes. Where changes are determined necessary, this content identifies the modifications made to the underlying criteria to derive the KP-FHR PDC. Wording removed is shown in **red** text with a strikethrough and wording added is shown in **blue** text with underline.

Basis: This content provides the justification and rationale for the final KP-FHR PDC.

Source: This content identifies the origin of the design criteria that is evaluated, and which informed the basis of review for the KP-FHR.

3.4 CONCLUSIONS

Kairos Power performed a comprehensive review of the Advanced Reactor Design Criteria in RG 1.232 and developed 46 PDC that meet the underlying safety objectives of 10 CFR 50 Appendix A. These PDC reflect the key design features of the KP-FHR technology and provide an appropriate set of requirements to facilitate the design and licensing of the KP-FHR. As such, these PDC apply for use by future license applicants under either 10 CFR 50 or 10 CFR 52 as long as the details of the KP-FHR progresses consistent with the key design attributes identified in Section 1.1.2.

4 REFERENCES

1. US Nuclear Regulatory Commission, "Guidance for Developing Principal Design Criteria for Non-Light Water Reactors," RG 1.232, Revision 0.
2. University of California Berkley Nuclear Engineering. 2015. *Fluoride Salt Cooled High Temperature Reactor*. [ONLINE] Available at: http://fhr.nuc.berkeley.edu. [Accessed 26 June 2018].
3. Kairos Power, LLC, "Design Overview of the Kairos Power Fluoride Salt Cooled, High Temperature Reactor (KP-FHR)", November 2018.
4. US Nuclear Regulatory Commission, "Preapplication Safety Evaluation Report for Power Reactor Innovative Small Modular (PRISM) Liquid-Metal Reactor," NUREG-1368, Final Report, February 1994.
5. Nuclear Energy Institute, "Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development," NEI 18-04 (Draft).
6. Nuclear Regulatory Commission, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," SECY-94-084, March 1994.

Table 1. Comparison of Advanced Reactor Designs

Design Attribute or Issue	Sodium Fast Reactor	Modular High Temperature Gas-Cooled Reactor	Kairos Power Fluoride-Cooled High Temperature Reactor
Core Design	Oxide or metal fuel with metal cladding	Fully ceramic pebble fuel	Fully ceramic pebble fuel
Power Density	Very High	Low	Intermediate
Coolant	Molten sodium	Helium gas	Molten salt (Flibe)
Coolant Chemical Activity	High (sodium/water or air)	Low (graphite/air)	Very low
Operating Pressure	Near Atmospheric	High	Near Atmospheric
Solid fission product mobility following fuel damage	Low (retained in sodium)	Low (retention in fuel)	Extremely low (retention in fuel and in Flibe coolant)
Containment Concept	Pressure-retaining, low-leakage containment	Functional containment	Functional containment
Decay Heat Removal	Passive	Passive	Passive
Intermediate Loop (between primary and power conversion)	Yes	No	Yes

Table 2. Summary of RG 1.232 Criteria Applicable to KP-FHR

Section	Total # Criteria	Applicable Criteria (adopted as is or modified)			Not Applicable	Comments
		ARDC	SFR	MHTGR		
Overall	5	5	0	0	0	Essentially same as ARDC except for changing “important to safety” to “safety significant”
Multiple Barriers	10	7	0	3	0	Changes made to adopt SARRDL, functional containment, and “safety significant” instead of “important to safety.”
Reactivity Control	9	7	0	2	0	Changes to adopt SARRDLs and “safety significant.”
Fluid Systems/Heat Transport Systems	17	8	0	3	6	Pressure-retaining containment building specific criteria not applicable due to KP-FHR functional containment
Reactor Containment	8	0	0	0	8	Pressure-retaining containment building specific criteria not applicable due to KP-FHR functional containment
Fuel and Radioactivity Control	5	4	0	1	0	Minor changes to reflect safety significance instead of important to safety
Additional SFR-DC	10	NA	3	NA	7	Three of the ten are adopted
Additional MHTGR-DC	3	NA	NA	3	0	All three MHTGR-DC are adopted to maintain geometry for passive heat removal
TOTAL	**67**	**31**	**3**	**12**	**21**	

NA – Not Applicable

Table 3. Modifications Made to ARDC, SFR-DC, and MHTGR-DC

Modification	PDC
Replacement of SAFDL with SARRDL	12, 17, 26, 33, 34, 73
Functional containment	16, 75
Replacement of "Important to Safety" with "Safety Significant"	1, 2, 3, 4, 5, 16, 17, 18, 20, 44, 61, 71, 73
Replacement of "Cold Shutdown" with "Safe Shutdown"	5, 19
Removal of reference to helium coolant	13, 75
Removal of reference to sodium	70, 71
Reactor coolant boundary quality standards commensurate with the safety significance	13, 14, 15, 28, 30, 31, 32, 33, 34
Replacement of primary coolant boundary with reactor coolant boundary	70, 73

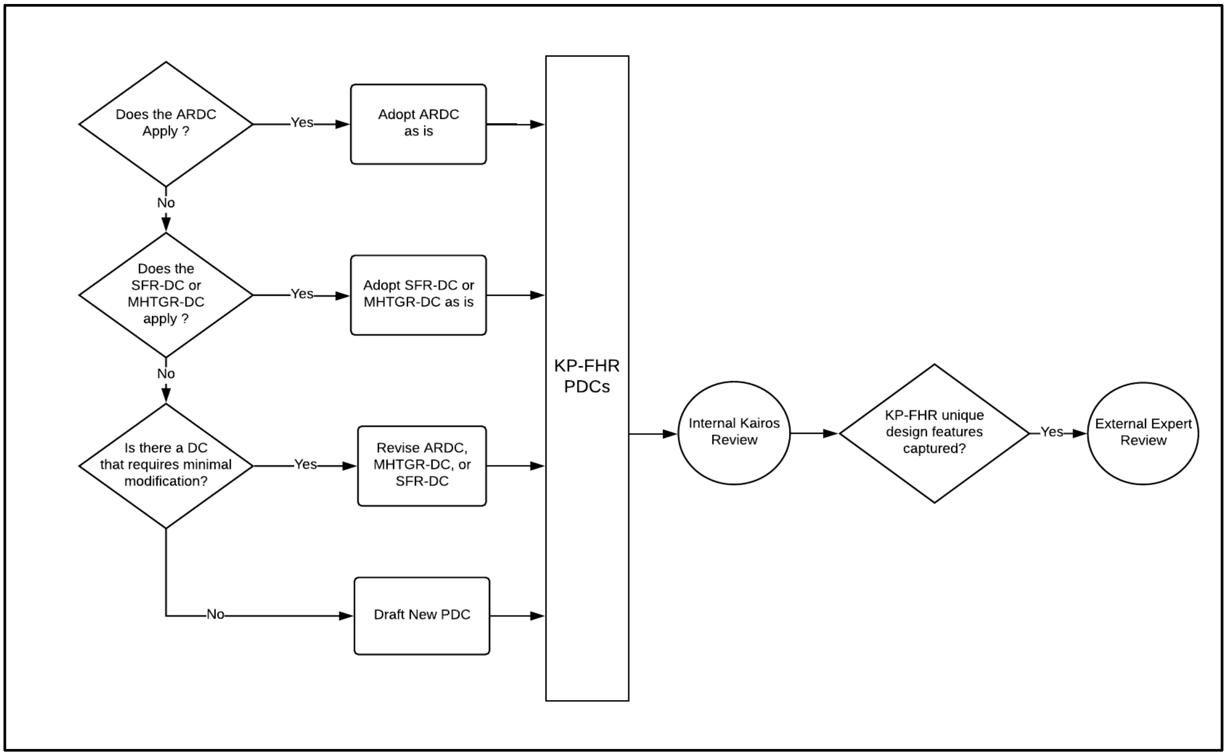

Figure 1. Flow Chart of PDC Development Methodology

APPENDIX A. PDC FOR THE KP-FHR

I. Overall Requirements

<table>
<tr><td>Title:</td><td colspan="2">1: Quality standards and records.</td></tr>
<tr><td>KP-FHR PDC:</td><td colspan="2">Structures, systems, and components which are safety significant shall be designed, fabricated, erected, and tested to quality standards commensurate with the safety significance of the 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 which are safety significant shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.</td></tr>
<tr><td rowspan="3">Position:</td><td colspan="2">The KP-FHR PDC 1 adopts the language from ARDC 1 of RG 1.232, Appendix A with the changes shown below</td></tr>
<tr><td>RG1.232, Appendix A, Criterion 1</td><td>KP-FHR PDC 1</td></tr>
<tr><td>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 significant shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.</td><td>Structures, systems, and components ~~important to safety~~ which are safety significant shall be designed, fabricated, erected, and tested to quality standards commensurate with the ~~importance of the safety~~ safety significance of the 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~~ which are safety significant shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.</td></tr>
</table>

Basis:	The ARDC is non-technology specific and therefore is applicable to the KP-FHR. The ARDC is modified to replace the term "important to safety" with "safety significant." The NRC staff defines "important to safety" generally in 10 CFR 50 Appendix A as "structures, systems, and components [SSCs] that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public." RG-1.232 provides guidance on development of PDC in lieu of Appendix A GDC. In an effort to clarify the scope of SSCs that perform risk-significant functions (i.e., safety related or non-safety-related but warranting special treatment), including those that perform functions that are necessary to meet defense-in-depth criteria, NEI-18-04 (Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development) explicitly defines the term "safety significant." The use of this term is endorsed in NRC's DG-1353, and its use herein is judged to describe more consistently the SSCs that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. Accordingly, "safety significant" is used in defining KP-FHR PDC in lieu of "important to safety" as used in Appendix A for GDC. The SFR-DC and MHTGR-DC are the same as the ARDC.
Source:	RG 1.232, Appendix A, Criterion 1

<table>
<tr><td>Title:</td><td colspan="2">2: Design bases for protection against natural phenomena.</td></tr>
<tr><td>KP-FHR PDC:</td><td colspan="2">Structures, systems, and components which are safety significant 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 safety significance of the functions to be performed.</td></tr>
<tr><td rowspan="3">Position:</td><td colspan="2">The KP-FHR PDC 2 adopts the language from ARDC 2 of RG 1.232, Appendix A with the modifications shown below.</td></tr>
<tr><td>RG 1.232, Appendix A, Criterion 2</td><td>KP-FHR PDC 2</td></tr>
<tr><td>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.</td><td>Structures, systems, and components ~~important to safety~~ which are safety significant 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~~ safety significance of the functions to be performed.</td></tr>
<tr><td>Basis:</td><td colspan="2">The ARDC is non-technology specific and therefore is applicable to the KP-FHR. The basis for the change to “importance to safety” is the same as in KP-FHR PDC 1.

The SFR-DC and MHTGR-DC are the same as the ARDC.</td></tr>
<tr><td>Source:</td><td colspan="2">RG 1.232, Appendix A, Criterion 2</td></tr>
</table>

<table>
<tr><td>Title:</td><td colspan="2">3: Fire protection.</td></tr>
<tr><td>KP-FHR PDC:</td><td colspan="2">Structures, systems, and components which are safety significant shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and fire-resistant materials shall be used wherever practical throughout the unit, particularly in locations with structures, systems, or components which are safety significant. 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 which are safety significant. 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.</td></tr>
<tr><td rowspan="3">Position:</td><td colspan="2">The KP-FHR PDC 3 adopts the language from ARDC 3 of RG 1.232, Appendix A with the modifications shown below.</td></tr>
<tr><td>RG 1.232, Appendix A, Criterion 3</td><td>KP-FHR PDC 3</td></tr>
<tr><td>Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and fire-resistant materials shall be used wherever practical throughout the unit, particularly in locations with structures, systems, or components important to safety. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to ensure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.</td><td>Structures, systems, and components which are ~~important to safety~~ safety significant shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and fire-resistant materials shall be used wherever practical throughout the unit, particularly in locations with structures, systems, or components ~~important to safety~~ which are safety significant. 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~~ which are safety significant. 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.</td></tr>
<tr><td>Basis:</td><td colspan="2">The ARDC is non-technology specific and therefore is applicable to the KP-FHR. The basis for the change to “important to safety” is provided in KP-FHR PDC 1.

The SFR-DC and MHTGR-DC are the same as the ARDC.</td></tr>
<tr><td>Source:</td><td colspan="2">RG 1.232, Appendix A, Criterion 3</td></tr>
</table>

Title:	4: Environmental and dynamic effects design bases.	
KP-FHR PDC:	Structures, systems, and components which are safety significant shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. These structures, systems, and components, shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.	
Position:	The KP-FHR PDC 4 adopts the language from ARDC 4 of RG 1.232, Appendix A with one modification as shown below.	
	RG 1.232, Appendix A, Criterion 4	KP-FHR 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, 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.	Structures, systems, and components ~~important to safety~~ which are safety significant shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. These structures, systems, and components, shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.
Basis:	The ARDC is sufficiently broad to include a wide range of advanced reactor technologies including KP-FHR. The basis for the change to “important to safety” is provided in KP-FHR PDC 1. MHTGR-DC is not adopted because it includes wording specific to features of MHTGR involving the potential of placing the turbine inside the helium pressure boundary, which does not apply to KP-FHR. The SFR-DC is not adopted because it added wording related to “liquid sodium” which is not present in the KP-FHR. This PDC also ensures that the environmental and dynamic effects from failures or events in non-safety significant systems (for example, steam generator tube rupture releasing high pressure fluids into interfacing systems) do not impact the ability of the safety-signficant SSCs to perform their safety functions.	

Source:	RG 1.232, Appendix A, Criterion 4

<table>
<tr><td>Title:</td><td colspan="2">5: Sharing of structures, systems, and components.</td></tr>
<tr><td>KP-FHR PDC:</td><td colspan="2">Structures, systems, and components which are safety significant 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, the ability to achieve and maintain safe shutdown of the remaining units.</td></tr>
<tr><td rowspan="3">Position:</td><td colspan="2">The KP-FHR PDC 5 adopts the language from ARDC 5 of RG 1.232, Appendix A with the modifications shown below.</td></tr>
<tr><td>RG 1.232, Appendix A, Criterion 5</td><td>KP-FHR PDC 5</td></tr>
<tr><td>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.</td><td>Structures, systems, and components ~~important to safety~~ which are safety significant 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, the ability to achieve ~~an orderly shutdown and cooldown~~ and maintain safe shutdown of the remaining units.</td></tr>
<tr><td>Basis:</td><td colspan="2">The ARDC is non-technology specific and therefore is applicable to the KP-FHR.
The basis for the modification to “important to safety” is provided in KP-FHR PDC 1.
The modification to the ARDC is made to replace the phrase “an orderly shutdown and cooldown” with “the ability to achieve and maintain safe shutdown”. This modification is made to remove the implied temperature of “cooldown.” The KP-FHR is a high-temperature system, with the reactor coolant melting temperature significantly above ambient temperatures. RG 1.232 Appendix A Criterion 26 rationale and SECY-94-084 describe a “safe shutdown” condition as: reactor subcriticality, decay heat removal, and radioactive materials containment. This modification is made to adopt this terminology.

The SFR-DC and MHTGR-DC are the same as the ARDC.</td></tr>
<tr><td>Source:</td><td colspan="2">RG 1.232, Appendix A, Criterion 5</td></tr>
</table>

II. Multiple Barriers

Title:	10: Reactor design.	
KP-FHR PDC:	The reactor core 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.	
Position:	The KP-FHR PDC 10 adopts the language from MHTGR-DC 10 of RG 1.232, Appendix C with one modification as discussed below.	
	RG 1.232, Appendix C, Criterion 10	KP-FHR 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 are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.	The reactor core ~~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.
Basis:	The ARDC is not adopted because it refers to SAFDLs, which are not appropriate for the KP-FHR, as explained below. The MHTGR-DC is adopted because the KP-FHR uses TRISO fuel particles which are used in the MHTGR along with a functional containment approach as described in KP-FHR PDC 16. The TRISO fuel particle is the primary fission product barrier and is expected to have a very low incremental fission product release during normal operation and AOOs. The MHTGR-DC uses the term SARRDL in place of SAFDL. The concept of SARRDL replacing SAFDL is discussed in RG 1.232, Appendix C for the MHTGR. SARRDLs limit the amount of radioactive inventory that is released by the system under normal operation and AOOs and is an indicator of functional containment integrity. The phrase "reactor core" is used (as in the ARDC) as opposed to "reactor system" in the MHTGR. The reactor core is the region within the reactor vessel where the fuel pebbles are critical during operation. For the MHTGR, the basis for SARRDLs was based on the depressurization of the system and the ejection of the entire circulating and plated out inventory. As such, in the MHTGR, the term "reactor system" refers to the fuel, helium coolant, and any connected systems that are not isolated and could contribute to dose. Although SARRDL has been determined to be the appropriate figure of merit for KP-FHR, the KP-FHR has a different basis for defining SARRDLs. Additionally, because the fission products are retained within the coolant, a loss of single phase, low-pressure coolant does not have the same radiological consequences as it does in an MHTGR. The SARRDLs are established so that the most limiting design basis accident does not exceed the siting regulatory dose limits criteria at the EAB and low-population zone, and also so that the 10 CFR 20.1301 annualized dose limits to the public are not exceeded at the EAB for normal operation and AOOs. The SFR-DC is not adopted because it uses a different fuel form than TRISO.	
Source:	RG 1.232, Appendix C, Criterion 10	

Title:	11: Reactor inherent protection.
KP-FHR PDC:	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.
Position:	The KP-FHR PDC 11 adopts the language from ARDC 11 of RG 1.232, Appendix A with no modifications.
Basis:	The ARDC is sufficiently broad to include a wide range of advanced reactor technologies including KP-FHR. The language in this ARDC is applicable and requires no KP-FHR specific modification. The MHTGR-DC and SFR-DC are the same as the ARDC.
Source:	RG 1.232, Appendix A, Criterion 11

Title:	12: Suppression of reactor power oscillations.	
KP-FHR PDC:	The reactor core; associated structures; and associated coolant, control, and protection systems shall be designed to ensure that power oscillations that can result in conditions exceeding specified acceptable system radionuclide release design limits are not possible or can be reliably and readily detected and suppressed.	
Position:	The KP-FHR PDC 12 adopts the language from ARDC 12 of RG 1.232, Appendix A with the modification shown below.	
	RG 1.232, Appendix A, Criterion 12	KP-FHR PDC 12
	The reactor core, associated structures, and associated coolant, control, and protection systems shall be designed to ensure that power oscillations that can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.	The reactor core, associated structures, and associated coolant, control, and protection systems shall be designed to ensure that power oscillations that can result in conditions exceeding specified acceptable ~~fuel~~ system radionuclide release design limits are not possible or can be reliably and readily detected and suppressed.
Basis:	The ARDC is sufficiently broad to include a wide range of advanced reactor technologies including KP-FHR. The ARDC is modified because it uses the term SAFDL. The basis for the use of SARRDL for KP-FHR is discussed in KP-FHR PDC 10. MHTGR-DC is not adopted because it states that "helium coolant" cannot impact stability, which is not the case for the molten salt coolant in the KP-FHR. The SFR-DC is the same as the ARDC.	
Source:	RG 1.232, Appendix A, Criterion 12	

Title:	13: Instrumentation and control.
KP-FHR PDC:	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, safety significant elements of the reactor coolant boundary, and functional containment. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.
Position:	The KP-FHR PDC 13 adopts the language from MHTGR-DC 13 of RG 1.232, Appendix C with the modifications shown below. **RG 1.232, Appendix C, Criterion 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. **KP-FHR 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, safety significant elements of the reactor ~~helium pressure~~coolant boundary, and functional containment. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.
Basis:	The ARDC is not adopted because it is based on a containment structure, and the KP-FHR uses a functional containment design. The MHTGR-DC is adopted because KP-FHR uses a functional containment design concept instead of a containment structure, which is similar to the MHTGR. The basis for the use of a functional containment in the KP-FHR is discussed in KP-FHR PDC 16. The addition of "safety significant" to the reactor coolant boundary is defined in the basis for the KP-FHR PDC 14. One additional modification is made to the MHTGR-DC to remove reference to "helium pressure" because the KP-FHR uses molten salt as its reactor coolant instead of helium. The KP-FHR is also near atmospheric pressure. The SFR-DC is not adopted because it does not use a functional containment.
Source:	RG 1.232, Appendix C, Criterion 13

Title:	14: Reactor coolant boundary.
KP-FHR PDC:	The safety significant elements of the reactor coolant boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and gross rupture.
Position:	The KP-FHR PDC 14 adopts the language from ARDC 14 of RG 1.232, Appendix A with the modification shown below.

RG 1.232, Appendix A, Criterion 14	KP-FHR PDC 14
The reactor coolant boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.	The safety significant elements of the reactor coolant boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

Basis:	The ARDC is sufficiently broad to include a wide range of advanced reactor technologies including KP-FHR. The reason for this modification is based on the consequence of leakage as related to the location of leaks or breaks within the systems and components containing reactor coolant system. The KP-FHR utilizes a low-pressure, inert reactor coolant as well as a low-pressure, inert cover gas. For the KP-FHR, the reactor coolant boundary is not a fission product barrier as is the case with traditional LWRs on which the original GDC was based. The KP-FHR reactor coolant boundary is defined as all those reactor components connected to and including the reactor vessel and the first physical interface or isolable geometry which separates the reactor coolant and cover gas from other systems. Certain components of the reactor coolant boundary (for example, those below a level which would expose the fuel) have a greater safety significance than other components, and the quality standards and design requirements should reflect this. These safety significant components of the reactor coolant boundary will be subject to leakage monitoring. The MHTGR-DC is not adopted because of the use of "helium-pressure boundary" and reference to a high pressure system where leakage anywhere can remove the coolant inventory from the system. The SFR-DC is not adopted because adding “primary” to “reactor coolant boundary” is not needed.
Source:	RG 1.232, Appendix A, Criterion 14

<table>
<tr><td>Title:</td><td colspan="2">15: Reactor coolant system design.</td></tr>
<tr><td>KP-FHR PDC:</td><td colspan="2">The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to ensure that the design conditions of the safety significant elements of the reactor coolant boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.</td></tr>
<tr><td rowspan="3">Position:</td><td colspan="2">The KP-FHR PDC 15 adopts the language from ARDC 15 of RG 1.232, Appendix A with one modification.</td></tr>
<tr><td>RG 1.232, Appendix A, Criterion 15</td><td>KP-FHR PDC 15</td></tr>
<tr><td>The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to ensure that the design conditions of the reactor coolant boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.</td><td>The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to ensure that the design conditions of the safety significant elements of the reactor coolant boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.</td></tr>
<tr><td>Basis:</td><td colspan="2">The addition of “safety significant” to reactor coolant boundary is added to align PDC 15 with PDC 14 and the basis for this change is given in the basis for the KP-FHR PDC 14.
The MHTGR is not adopted because it uses helium pressure boundary. The SFR-DC is not adopted because adding “primary” to “reactor coolant boundary” is not needed.</td></tr>
<tr><td>Source:</td><td colspan="2">RG 1.232, Appendix A, Criterion 15</td></tr>
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<table>
<tr><td>Title:</td><td colspan="2">16: Containment design.</td></tr>
<tr><td>KP-FHR PDC:</td><td colspan="2">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 which are safety significant are not exceeded for as long as postulated accident conditions require.</td></tr>
<tr><td rowspan="3">Position:</td><td colspan="2">The KP-FHR PDC 16 adopts the language from MHTGR-DC 16 of RG 1.232, Appendix C with one modification as shown below.</td></tr>
<tr><td>RG 1.232, Appendix C, Criterion 16</td><td>KP-FHR PDC 16</td></tr>
<tr><td>A reactor functional containment, consisting of multiple barriers internal and/or external to thereactor and its cooling system, shall be provided to control the release of radioactivity to the environment and to ensure that the functional containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.</td><td>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 which are safety significant ~~important to safety~~ are not exceeded for as long as postulated accident conditions require.</td></tr>
<tr><td>Basis:</td><td colspan="2">The ARDC is not adopted because it is based on a containment structure, and the KP-FHR uses a functional containment design.

The MHTGR-DC is adopted because the KP-FHR design relies on a functional containment approach similar to the Modular High Temperature Gas Reactor (MHTGR) instead of the typical LWR pressure-retaining containment structure. The KP-FHR functional containment safety design objective is to meet 10 CFR 50.34 (10 CFR 52.79) offsite dose requirements at the plant's EAB with margin. A functional containment is defined in RG 1.232 as a "barrier, or set of barriers taken together, that effectively limit the physical transport and release of radionuclides to the environment across a full range of normal operating conditions, AOOs, and accident conditions." The functional containment controls radionuclides at their source within the multiple layers of the coated TRISO fuel particle without requiring active design features or operator actions to ensure that the dose at the site boundary as a consequence of postulated accidents meets regulatory limits. Additionally, the molten salt coolant serves as an additional distinct barrier providing retention of fission products that escape the fuel particle and fuel pebble barriers.

The basis for the replacement of “important to safety” with “safety significant” is given in the basis for KP-FHR PDC 1.

The SFR-DC is not adopted because it is not based on a functional containment design.</td></tr>
<tr><td>Source:</td><td colspan="2">RG 1.232, Appendix C, Criterion 16</td></tr>
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<table>
<tr><td>Title:</td><td colspan="2">17: Electric power systems.</td></tr>
<tr><td>KP-FHR PDC:</td><td colspan="2">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 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 safety significant functions is provided.</td></tr>
<tr><td rowspan="3">Position:</td><td colspan="2">The KP-FHR PDC 17 adopts the language from ARDC 17 of RG 1.232, Appendix C with the modifications shown below.</td></tr>
<tr><td>RG 1.232, Appendix A, Criterion 17</td><td>KP-FHR PDC 17</td></tr>
<tr><td>Electric power systems shall be provided when required to permit functioning of structures, systems, and components. The safety function for each power system shall be to provide sufficient capacity and capability to ensure that
(1) that the design limits for the fission product barriers are not exceeded as a result of anticipated operational occurrences and
(2) safety functions that rely on electric power are maintained in the event of postulated accidents.

The electric power systems shall include an onsite power system and an additional power system. The onsite electric power system shall have sufficient independence, redundancy, and testability to perform its safety functions, assuming a single failure. An additional power system shall have sufficient independence and testability to perform its safety function.

If electric power is not needed for anticipated operational occurrences or postulated accidents, the design shall demonstrate that power for important to safety functions is provided.</td><td>Electric power systems shall be provided when required to permit functioning of structures, systems, and components. The safety function for each power system shall be to provide sufficient capacity and capability to ensure that
(1) that the ~~design limits for the fission product barriers~~ specified acceptable system radionuclide release 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</td></tr>
</table>

		power for ~~important to safety~~ safety significant functions is provided.
Basis:	The ARDC is sufficiently broad to include a wide range of advanced reactor technologies including KP-FHR. The language in this ARDC requires two modifications. The term "design limits for fission product barriers" is changed to "specified acceptable system radionuclide release design limits (SARRDL)" to be consistent with the KP-FHR figure of merit as described in the basis for PDC 10. The basis for the replacement of "important to safety" with "safety significant" is given in KP-FHR PDC 1. As discussed in the ARDC, this use of two phrases ("safety functions" and "important to safety functions") is intentional to imply two different terms and sets of functions. "Safety functions" (in the context of power supplies) means power for safety-related SSCs; "safety-significant functions" used in the PDC (replacing "important to safety functions") means "safety-significant functions not credited for design basis accidents", i.e. power for monitoring, lighting, etc. This second class of functions is not required to be Class-1E and is not required to be redundant. The MHTGR-DC is not adopted because it uses the words "helium pressure boundary," which does not apply to the KP-FHR. The SFR-DC is the same as the ARDC.	
Source:	RG 1.232, Appendix A, Criterion 17	

<table>
<tr><td>Title:</td><td colspan="2">18: Inspection and testing of electric power systems.</td></tr>
<tr><td>KP-FHR PDC:</td><td colspan="2">Electric power systems which are safety significant 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.</td></tr>
<tr><td rowspan="3">Position:</td><td colspan="2">The KP-FHR PDC 18 adopts the language from ARDC 18 of RG 1.232, Appendix A with one modification as described below.</td></tr>
<tr><td>RG 1.232, Appendix A, Criterion 18</td><td>KP-FHR PDC 18</td></tr>
<tr><td>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.</td><td>Electric power systems ~~important to safety~~ which are safety significant 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.</td></tr>
<tr><td>Basis:</td><td colspan="2">The ARDC is design independent and sufficiently broad to include a wide range of advanced reactor technologies including KP-FHR. The language in this ARDC requires one modification. The basis for the change to “important to safety” is provided in KP-FHR PDC 1.

The MHTGR-DC and SFR-DC are the same as the ARDC.</td></tr>
<tr><td>Source:</td><td colspan="2">RG 1.232, Appendix A, Criterion 18</td></tr>
</table>

<table>
<tr><td>Title:</td><td colspan="2">19: Control room.</td></tr>
<tr><td>KP-FHR PDC:</td><td colspan="2">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 shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during shutdown, and (2) with a potential capability for subsequent safe shutdown of the reactor through the use of suitable procedures.</td></tr>
<tr><td rowspan="3">Position:</td><td colspan="2">The KP-FHR PDC 19 adopts the language from ARDC 19 of RG 1.232, Appendix A with the modifications below.</td></tr>
<tr><td>RG 1.232, Appendix A, Criterion 19</td><td>KP-FHR PDC 19</td></tr>
<tr><td>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.</td><td>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~~ safe shutdown of the reactor through the use of suitable procedures.</td></tr>
</table>

Basis:	The ARDC is sufficiently broad to include a wide range of advanced reactor technologies including KP-FHR. As a design objective, there are no operator actions anticipated to be required to safely shut down the reactor and maintain it in a safe state for the KP-FHR. The ARDC includes the word “maintain” in the first paragraph, and the KP-FHR PDC retains this terminology. However, if the design demonstrates that required safety systems respond automatically such that the operator has no required actions and only needs to monitor the reactor during an accident, this meets the language in the criteria for “maintain.” There is no requirement implied by the use of “maintain” that the criteria could only be met by inclusion of active operator actions to ensure a safe condition. The basis for the modification to change from “cold shutdown” to “safe shutdown” is given in the basis for KP-FHR PDC 5. The MHTGR-DC is the same as the ARDC. The SFR-DC is not adopted because it contains language related to “sodium aerosols,” and KP-FHR does not use sodium in its design.
Source:	RG 1.232, Appendix A, Criterion 19

III. Reactivity Control

Title:	20: Protection system functions.
KP-FHR PDC:	The protection system shall be designed (1) to initiate automatically the operation of appropriate systems, including the reactivity control systems, to assure that specified acceptable system radionuclide release design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components which are safety significant.
Position:	The KP-FHR PDC 20 adopts the language from MHTGR-DC 20 of RG 1.232, Appendix C with the modifications shown below. **RG 1.232, Appendix C, Criterion 20:** The protection system shall be designed (1) to initiate automatically the operation of appropriate systems, including the reactivity control systems, to assure that specified acceptable 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. **KP-FHR PDC:** The protection system shall be designed (1) to initiate automatically the operation of appropriate systems, including the reactivity control systems, to assure that specified acceptable system radionuclide release design limits ~~is~~ are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components ~~important to safety~~which are safety significant.
Basis:	The ARDC is not adopted because it is based on the use of SAFDLs which as discussed in KP-FHR PDC 10, is being replaced by SARRDLs. The basis for the use SARRDLs instead of SAFDLs is provided in KP-FHR PDC 10. The first change corrects a grammatical error in the MHTGR-DC. The basis for the modification to replace “important to safety” with “safety significant” is given in KP-FHR PDC 1. SFR-DC 10 is not adopted because it also uses SAFDLs.
Source:	RG 1.232, Appendix C, Criterion 20

Title:	21: Protection system reliability and testability.
KP-FHR PDC:	The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.
Position:	The KP-FHR PDC 21 adopts the language from ARDC 21 of RG 1.232, Appendix A with no modifications.
Basis:	The ARDC is sufficiently broad to include a wide range of advanced reactor technologies including KP-FHR. The language in this ARDC, which adopts the GDC with no changes, is applicable and requires no KP-FHR specific modification. The MHTGR-DC and SFR-DC are both the same as the ARDC.
Source:	RG 1.232, Appendix A, Criterion 21

Title:	22: Protection system independence.
KP-FHR PDC:	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.
Position:	The KP-FHR PDC 22 adopts the language from ARDC 22 of RG 1.232, Appendix A with no modifications.
Basis:	The ARDC is sufficiently broad to include a wide range of advanced reactor technologies including KP-FHR. The language in this ARDC, which adopts the GDC with no changes, is applicable and requires no KP-FHR specific modification. The MHTGR-DC and SFR-DC are the same as the ARDC.
Source:	RG 1.232, Appendix A, Criterion 22

Title:	23: Protection system failure modes.
KP-FHR PDC:	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.
Position:	The KP-FHR PDC 23 adopts the language from ARDC 23 of 10 CFR 50, Appendix A with no modifications.
Basis:	The ARDC is sufficiently broad to include a wide range of advanced reactor technologies including KP-FHR. The language in this ARDC, which adopts the GDC with no changes, is applicable and requires no KP-FHR specific modification. The SFR-DC does not apply because it refers to “sodium” which is not present in the KP-FHR. The MHTGR-DC is the same as the ARDC.
Source:	RG 1.232, Appendix A, Criterion 23

Title:	24: Separation of protection and control systems.
KP-FHR PDC:	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.
Position:	The KP-FHR PDC 24 adopts the language from ARDC 24 of RG 1.232, Appendix A with no modifications.
Basis:	The ARDC is sufficiently broad to include a wide range of advanced reactor technologies including KP-FHR. The language in this ARDC, which adopts the GDC with no changes, is applicable and requires no KP-FHR specific modification. The MHTGR-DC and SFR-DC are both the same as the ARDC.
Source:	RG 1.232, Appendix A, Criterion 24

Title:	25: Protection system requirements for reactivity control malfunctions.
KP-FHR PDC:	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.
Position:	The KP-FHR PDC 25 adopts the language from MHTGR-DC 25 of RG 1.232 Appendix C with no modifications.
Basis:	The ARDC is not adopted because it refers to SAFDLs. The MHTGR-DC is adopted because it is based on the use of SARRDL. The basis for the use of SARRDL instead of SAFDL is provided in KP-FHR PDC 10. The SFR-DC is not adopted because it uses the term SAFDL.
Source:	RG 1.232, Appendix C, Criterion 25

Title:	26: Reactivity control systems.	
KP-FHR PDC:	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 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 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.	
Position:	The KP-FHR PDC 26 adopts the language from ARDC 26 of RG 1.232, Appendix A with one modification.	
	RG 1.232, Appendix A, Criterion 26	KP-FHR PDC 26
	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	A minimum of two reactivity control systems or means shall provide: (1) A means of inserting negative reactivity at a sufficient rate and amount to assure, with

	appropriate margin for malfunctions, that the design limits for the fission product barriers are not exceeded and safe shutdown is achieved and maintained during normal operation, including anticipated operational occurrences. (2) A means which is independent and diverse from the other(s), shall be capable of controlling the rate of reactivity changes resulting from planned, normal power changes to assure that the design limits for the fission product barriers are not exceeded. (3) A means of inserting negative reactivity at a sufficient rate and amount to assure, with appropriate margin for malfunctions, that the capability to cool the core is maintained and a means of shutting down the reactor and maintaining, at a minimum, a safe shutdown condition following a postulated accident. (4) A means for holding the reactor shutdown under conditions which allow for interventions such as fuel loading, inspection and repair shall be provided.	appropriate margin for malfunctions, that the specified acceptable system radionuclide release design limits ~~design limits for the fission product barriers~~ are not exceeded and safe shutdown is achieved and maintained during normal operation, including anticipated operational occurrences. (2) A means which is independent and diverse from the other(s), shall be capable of controlling the rate of reactivity changes resulting from planned, normal power changes to assure that the ~~design limits for the fission product barriers~~ specified acceptable system radionuclide release 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.
Basis:	The ARDC is sufficiently broad to include a wide range of advanced reactor technologies including KP-FHR. The language in this ARDC is adopted with one modification. The “specified acceptable fuel design limit” is changed to “specified acceptable system radionuclide release design limits (SARRDL)” to be consistent with the KP-FHR figure of merit as described in the basis for PDC 10. The MHTGR-DC is not adopted because it uses wording related to “helium pressure boundary.” The SFR-DC uses the same wording as the ARDC.	
Source:	RG 1.232, Appendix A, Criterion 26	

Title:	28: Reactivity limits.	
KP-FHR PDC:	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 safety significant elements of the reactor coolant boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures, or other reactor vessel internals to impair significantly the capability to cool the core.	
Position:	The KP-FHR PDC 28 adopts the language from ARDC 28 of RG 1.232, Appendix A with one modification.	
	RG 1.232, Appendix A, Criterion 28	KP-FHR PDC 28
	The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to ensure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures, or other reactor vessel internals to impair significantly the capability to cool the core.	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 safety significant elements of the reactor coolant boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures, or other reactor vessel internals to impair significantly the capability to cool the core.
Basis:	The ARDC is sufficiently broad to include a wide range of advanced reactor technologies including KP-FHR. The addition of “safety significant” to reactor coolant boundary is added to align this PDC with PDC 14 and PDC 15 and the basis for this change is given in the basis for the KP-FHR PDC 14. The reactor coolant boundary is defined in the basis for KP-FHR PDC 14. The MHTGR-DC is not adopted because it mentions "helium pressure boundary". The SFR-DC adds “primary” to “reactor coolant boundary,” a distinction which is not needed for KP-FHR.	
Source:	RG 1.232, Appendix A, Criterion 28	

Title:	29: Protection against anticipated operational occurrences.
KP-FHR PDC:	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.
Position:	The KP-FHR PDC 29 adopts the language from ARDC 29 of RG 1.232, Appendix A with no modifications.
Basis:	The ARDC is sufficiently broad to include a wide range of advanced reactor technologies including KP-FHR. The language in this ARDC, which adopts the GDC with no changes, is applicable and requires no KP-FHR specific modification. The MHTGR-DC and SFR-DC are both the same as the ARDC.
Source:	RG 1.232, Appendix A, Criterion 29

IV. Heat Transport System

Title:	30: Quality of reactor coolant boundary.	
KP-FHR PDC:	Components that are part of the reactor coolant boundary shall be designed, fabricated, erected, and tested to quality standards commensurate with the safety significance of the functions to be performed. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage from safety-significant elements of the reactor coolant boundary.	
Position:	The KP-FHR PDC 30 adopts the language from ARDC 30 of RG 1.232 Appendix A with the modifications shown below.	
	RG 1.232, Appendix A, Criterion 30	KP-FHR PDC 30
	Components that are part of the reactor coolant boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.	Components that are part of the reactor coolant boundary shall be designed, fabricated, erected, and tested to ~~the highest quality standards practical~~ quality standards commensurate with the safety significance of the functions to be performed. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage from safety-significant elements of the reactor coolant boundary.
Basis:	The ARDC is sufficiently broad to include a wide range of advanced reactor technologies including KP-FHR. The KP-FHR PDC adopt this ARDC with two modifications. The language "to the highest quality standards practical" is replaced with the language from KP-FHR PDC-1 (Quality standards and records), so that "the highest quality standards practical" reads "quality standards commensurate with the safety significance of the functions to be performed". This language is consistent with the language in GDC 1 and better reflects the safety significance of the reactor coolant boundary in the KP-FHR. The KP-FHR utilizes a low-pressure, inert reactor coolant as well as a low-pressure, inert cover gas. Certain components of the reactor coolant boundary (for example those could expose the fuel in the event of unmitigated leakage) have a greater significance than other components, and the quality standards should reflect this. The reactor coolant boundary is defined in the basis for KP-FHR PDC 14. The phrase “from safety-significant elements of the reactor coolant boundary” is added to the last sentence. The basis for this is given in PDC 14. The MHTGR-DC is not adopted because it refers to “helium” as a reactor coolant, which is not part of the KP-FHR design. The SFR-DC adds “primary” to “reactor coolant boundary,” a distinction which is not needed for KP-FHR.	
Source:	RG 1.232, Appendix A, Criterion 30	

Title:	31: Fracture prevention of reactor coolant boundary.	
KP-FHR PDC:	The safety significant elements of the reactor coolant boundary shall be designed with sufficient margin to ensure that when stressed under operating, maintenance, testing, and postulated accident conditions, (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures, service degradation of material properties, creep, fatigue, stress rupture, and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation and coolant composition, including contaminants and reaction products, on material properties, (3) residual, steady-state, and transient stresses, and (4) size of flaws.	
Position:	The KP-FHR PDC 31 adopts the language from ARDC 31 of RG 1.232, Appendix A with one modification as shown below.	
	RG 1.232, Appendix A, Criterion 31	KP-FHR PDC 31
	The reactor coolant boundary shall be designed with sufficient margin to ensure that when stressed under operating, maintenance, testing, and postulated accident conditions, (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures, service degradation of material properties, creep, fatigue, stress rupture, and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation and coolant composition, including contaminants and reaction products, on material properties, (3) residual, steady-state, and transient stresses, and (4) size of flaws.	The safety significant elements of the reactor coolant boundary shall be designed with sufficient margin to ensure that when stressed under operating, maintenance, testing, and postulated accident conditions, (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures, service degradation of material properties, creep, fatigue, stress rupture, and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation and coolant composition, including contaminants and reaction products, on material properties, (3) residual, steady-state, and transient stresses, and (4) size of flaws.
Basis:	The ARDC is sufficiently broad to include a wide range of advanced reactor technologies including KP-FHR. The basis for this modification to add the phrase “safety significant” to the PDC is given in KP-FHR PDC 14. The reactor coolant boundary is defined in the basis for KP-FHR PDC 14. The consequences of brittle failure are applied where they have safety significance. Many portions of the reactor coolant boundary will not need to be protected from non-brittle failure and rapidly propagating failure to ensure the safety of the plant. Additionally, this does not supercede the requirements in PDC 4 that the coolant itself cannot be allowed to cause failures in safety systems. For these reasons, the “safety significant” language was added. The safety-significance is not lowered, it is set to be variable and reflect the safety analysis. The MHTGR-DC is not adopted because the KP-FHR does not use helium as a coolant. The SFR-DC is not adopted because the ARDC covers the fracture prevention requirement sufficiently and adding “primary” to “reactor coolant boundary” is not needed.	

Source:	RG 1.232, Appendix A, Criterion 31

<table>
<tr><td>Title:</td><td>32: Inspection of reactor coolant boundary.</td></tr>
<tr><td>KP-FHR PDC:</td><td>Safety significant components that are part of the reactor coolant boundary shall be designed to permit (1) periodic inspection, monitoring, or functional testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor vessel.</td></tr>
<tr><td>Position:</td><td>The KP-FHR PDC 32 adopts the language from ARDC 32 of RG 1.232, Appendix A with two modifications.
<table>
<tr><td>RG 1.232, Appendix A, Criterion 32</td><td>KP-FHR PDC 32</td></tr>
<tr><td>Components that are part of the reactor coolant boundary shall be designed to permit (1) periodic inspection and functional testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor vessel.</td><td>Safety significant components that are part of the reactor coolant boundary shall be designed to permit (1) periodic inspection, monitoring, ~~and~~ or functional testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor vessel.</td></tr>
</table></td></tr>
<tr><td>Basis:</td><td>The ARDC is sufficiently broad to include a wide range of advanced reactor technologies including KP-FHR. The language in this ARDC is applicable with two modifications.
The phrase “monitoring, or” is added, as monitoring may be another means to accomplish the same objectives as are intended by the PDC while ensuring flexibility if inspections and functional testing are not feasible. The phrase “safety significant” is added to reactor coolant pressure boundary. The basis for this change is given in the basis for PDC 14. The reactor coolant boundary is defined in the basis for KP-FHR PDC 14.
PDC 35, 36, and 37 provide requirements to ensure the capability to cool the core is maintained. As such, the potential for flow blockages/restrictions from failed internals (such as graphite reflector blocks) is addressed as part of compliance with PDC 35, 36, and 37, including inspections if appropriate.
The MHTGR-DC is not adopted because it uses helium as a coolant. SFR-DC is not adopted because the ARDC covers the inspection requirement sufficiently and adding primary to reactor coolant boundary is not needed.</td></tr>
<tr><td>Source:</td><td>RG 1.232, Appendix A, Criterion 32</td></tr>
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<table>
<tr><td>Title:</td><td colspan="2">33: Reactor coolant inventory maintenance.</td></tr>
<tr><td>KP-FHR PDC:</td><td colspan="2">A system to maintain reactor coolant inventory for protection against small breaks in the safety significant elements of the reactor coolant boundary shall be provided as necessary to ensure that specified acceptable system radionuclide release design limits are not exceeded as a result of reactor coolant inventory loss due to leakage from the reactor coolant boundary and rupture of small piping or other small components that are part of the boundary. The system shall be designed to ensure that the system safety function can be accomplished using the piping, pumps, and valves used to maintain reactor coolant inventory during normal reactor operation.</td></tr>
<tr><td rowspan="3">Position:</td><td colspan="2">The KP-FHR PDC 33 adopts the language from ARDC 33 of RG 1.232, Appendix A with the modifications shown below.</td></tr>
<tr><td>RG 1.232, Appendix A, Criterion 33</td><td>KP-FHR PDC 33</td></tr>
<tr><td>A system to maintain reactor coolant inventory for protection against small breaks in the reactor coolant boundary shall be provided as necessary to ensure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant inventory loss due to leakage from the reactor coolant boundary and rupture of small piping or other small components that are part of the boundary. The system shall be designed to ensure that the system safety function can be accomplished using the piping, pumps, and valves used to maintain reactor coolant inventory during normal reactor operation.</td><td>A system to maintain reactor coolant inventory for protection against small breaks in the safety significant elements of the reactor coolant boundary shall be provided as necessary to ensure that specified acceptable ~~fuel~~ system radionuclide release design limits are not exceeded as a result of reactor coolant inventory loss due to leakage from the reactor coolant boundary and rupture of small piping or other small components that are part of the boundary. The system shall be designed to ensure that the system safety function can be accomplished using the piping, pumps, and valves used to maintain reactor coolant inventory during normal reactor operation.</td></tr>
<tr><td>Basis:</td><td colspan="2">The ARDC is sufficiently broad to include a wide range of advanced reactor technologies including KP-FHR. The basis for the change from SAFDL to SARRDL is provided in KP-FHR PDC 10. The phrase “safety significant elements” is added to reactor coolant boundary and the basis is given in the basis for PDC 14.
The SFR-DC is not adopted because of reference to SAFDLs. There is no MHTGR-DC for Reactor coolant inventory maintenance.</td></tr>
<tr><td>Source:</td><td colspan="2">RG 1.232, Appendix A, Criterion 33</td></tr>
</table>

<table>
<tr><td>Title:</td><td colspan="2">34: Residual heat removal.</td></tr>
<tr><td>KP-FHR PDC:</td><td colspan="2">A system to remove residual heat shall be provided. For normal operations and anticipated operational occurrences, the system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable system radionuclide release design limits and the design conditions of safety significant elements of the reactor coolant boundary are not exceeded.

Suitable redundancy in components and features and suitable interconnections, leak detection, and isolation capabilities shall be provided to ensure that the system safety function can be accomplished, assuming a single failure.</td></tr>
<tr><td rowspan="3">Position:</td><td colspan="2">The KP-FHR PDC 34 adopts the language from ARDC 34 of RG 1.232, Appendix A with the modifications shown below.</td></tr>
<tr><td>RG 1.232, Appendix A, Criterion 34</td><td>KP-FHR PDC 34</td></tr>
<tr><td>A system to remove residual heat shall be provided. For normal operations and anticipated operational occurrences, the system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant boundary are not exceeded.

Suitable redundancy in components and features and suitable interconnections, leak detection, and isolation capabilities shall be provided to ensure that the system safety function can be accomplished, assuming a single failure.</td><td>A system to remove residual heat shall be provided. For normal operations and anticipated operational occurrences, the system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable ~~fuel~~ system radionuclide release design limits and the design conditions of safety significant elements of the reactor coolant boundary are not exceeded.

Suitable redundancy in components and features and suitable interconnections, leak detection, and isolation capabilities shall be provided to ensure that the system safety function can be accomplished, assuming a single failure.</td></tr>
<tr><td>Basis:</td><td colspan="2">The KP-FHR includes two decay heat removal systems, one is a non-safety-related system for normal operation and AOOs and the second one is a safety-related passive decay heat removal system for accidents. The ARDC is adopted for modification because the KP-FHR includes multiple systems for decay heat removal consistent with the ARDC wording. One modification to the ARDC is made to change SAFDL to SARRDL because the KP-FHR has adopted that term as described in the basis for KP-FHR PDC 10. The reactor coolant boundary is defined in the basis for KP-FHR PDC 14. The basis for adding the phrase “safety significant elements” is given in the basis for PDC 14.
The SFR-DC is not adopted because it refers to SAFDLs. The MHTGR-DC is not adopted because it references a passive system.</td></tr>
<tr><td>Source:</td><td colspan="2">RG 1.232, Appendix A, Criterion 34</td></tr>
</table>

<table>
<tr><td>Title:</td><td colspan="2">35: Passive residual heat removal.</td></tr>
<tr><td>KP-FHR PDC:</td><td colspan="2">A system to assure sufficient core cooling during postulated accidents and to remove residual heat following postulated accidents shall be provided. The system safety function shall be to transfer heat from the reactor core during and following postulated accidents such that fuel and reactor internal structure damage that could interfere with continued effective core cooling is prevented.</td></tr>
<tr><td rowspan="3">Position:</td><td colspan="2">The KP-FHR PDC 35 adopts the language from ARDC 35 of RG 1.232, Appendix A with the modifications below.</td></tr>
<tr><td>RG 1.232, Appendix A, Criterion 35</td><td>KP-FHR PDC 35</td></tr>
<tr><td>Emergency core cooling system.

A system to assure sufficient core cooling during postulated accidents and to remove residual heat following postulated accidents shall be provided. The system safety function shall be to transfer heat from the reactor core during and following postulated accidents such that fuel and clad damage that could interfere with continued effective core cooling is prevented.</td><td>~~Emergency core cooling system~~ <u>Passive residual heat removal</u>.

A system to assure sufficient core cooling during postulated accidents and to remove residual heat following postulated accidents shall be provided. The system safety function shall be to transfer heat from the reactor core during and following postulated accidents such that fuel and ~~clad~~ <u>reactor internal structure</u> damage that could interfere with continued effective core cooling is prevented.</td></tr>
<tr><td>Basis:</td><td colspan="2">The ARDC is adopted for modification as described in the basis for PDC 34. One modification is made to the title of this PDC because the KP-FHR design does not include an Emergency Core Cooling System (ECCS) which injects coolant into the core. Another modification is made to remove the reference to clad damage, a term that does not apply to the KP-FHR. Clad is replaced with the phrase “reactor internal structure.” The design objective is that sufficient coolant circulation is established through the internals and fuel in concert with the operation of the residual heat removal system.
The use of the word “system” in this PDC includes those mechanical elements, flow paths, and features that support the function of residual heat removal.
The MHTGR-DC is not adopted because it calls this design criteria not applicable. The SFR-DC is the same as the ARDC.</td></tr>
<tr><td>Source:</td><td colspan="2">RG 1.232, Appendix A, Criterion 35</td></tr>
</table>

Title:	36: Inspection of passive residual heat removal system.
KP-FHR PDC:	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.
Position:	The KP-FHR PDC 36 adopts the language from and the title for MHTGR-DC 36 of RG 1.232, Appendix C with no modifications.
Basis:	The ARDC is not adopted for the KP-FHR because it refers to Emergency Core Cooling, which is a system the KP-FHR does not have or need. The basis for the adoption of the MHTGR-DC is that it refers to a passive residual heat removal system which applies to the KP-FHR. The SFR-DC is not adopted because it is the same as the ARDC.
Source:	RG 1.232, Appendix C, Criterion 36

Title:	37: Testing of passive residual heat removal system.
KP-FHR PDC:	The passive residual heat removal system shall be designed to permit appropriate periodic functional testing to ensure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the system components, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including associated systems, for AOO or postulated accident decay heat removal to the ultimate heat sink and, if applicable, any system(s) necessary to transition from active normal operation to passive mode.
Position:	The KP-FHR PDC 37 adopts the language from MHTGR-DC 37 of RG 1.232, Appendix C.
Basis:	The ARDC is not adopted because it refers to Emergency Core Cooling, which is a system the KP-FHR does not have or need. The basis for the adoption of the MHTGR-DC is the same as in the basis for KP-FHR PDC 36. The SFR-DC is not adopted because it is the same as the ARDC.
Source:	RG 1.232, Appendix C, Criterion 37

Title:	44: Structural and equipment cooling.	
KP-FHR PDC:	In addition to the heat rejection capability of the passive residual heat removal system, systems to transfer heat from structures, systems, and components which are safety significant 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.	
Position:	The KP-FHR PDC 44 adopts the language from MHTGR-DC 44 of RG 1.232, Appendix C with one change as shown below.	
	RG 1.232, Appendix C, Criterion 44	KP-FHR 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.	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~~ which are safety significant 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.
Basis:	The MHTGR-DC is adopted because the KP-FHR design relies on a passive residual heat removal system to remove post accident decay heat. The basis for the modification to "important to safety" is described in KP-FHR PDC 1. The SFR-DC is not adopted because it is the same as the ARDC.	
Source:	RG 1.232, Appendix C, Criterion 44	

Title:	45: Inspection of structural and equipment cooling systems.
KP-FHR PDC:	The structural and equipment cooling systems shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to ensure the integrity and capability of the systems.
Position:	The KP-FHR PDC 45 adopts the wording from ARDC 45 of RG 1.232, Appendix A with no modifications.
Basis:	The ARDC 45 is in direct suppport of ARDC 44. Although the MHTGR-DC 44 is adopted, the language in this ARDC is applicable and requires no KP-FHR specific modification. The SFR-DC and MHTGR-DC are the same as the ARDC.
Source:	RG 1.232, Appendix A, Criterion 45

Title:	46: Testing of structural and equipment cooling systems.
KP-FHR PDC:	The structural and equipment cooling systems shall be designed to permit appropriate periodic functional testing to ensure (1) the structural and leaktight integrity of their components, (2) the operability and performance of the system components, and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequences that bring the systems into operation for reactor shutdown and postulated accidents, including the operation of associated systems.
Position:	The KP-FHR PDC 46 adopts the language from ARDC 46 of RG 1.232, Appendix A with no modifications.
Basis:	The ARDC 46 is in direct suppport of ARDC 44. Although the MHTGR-DC 44 is adopted, the language in this ARDC is applicable and requires no KP-FHR specific modification. The SFR-DC 46 and MHTGR-DC 46 are the same as ARDC 46.
Source:	RG 1.232, Appendix A, Criterion 46

V. Reactor Containment

None.

VI. Fuel and Radioactivity

Title:	60 : Control of releases of radioactive materials to the environment.
KP-FHR PDC:	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.
Position:	The KP-FHR PDC 60 adopts the language from ARDC 60 of RG 1.232, Appendix A with no modifications.
Basis:	The ARDC is non-technology specific and therefore is applicable to the KP-FHR. The language in this ARDC, which adopts the GDC with no changes, is applicable and requires no KP-FHR specific modification. The SFR-DC and MHTGR-DC are the same as the ARDC.
Source:	RG 1.232, Appendix A, Criterion 60

Title:	61: Fuel storage and handling and radioactivity control.	
KP-FHR PDC:	The fuel storage and handling, radioactive waste, and other systems that may contain radioactivity shall be designed to ensure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components which are safety significant, (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 safety significance of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage cooling under accident conditions.	
Position:	The KP-FHR PDC 61 adopts the language from ARDC 61 of RG 1.232, Appendix A with the modifications shown below.	
	RG 1.232, Appendix A, Criterion 61	KP-FHR PDC 61
	The fuel storage and handling, radioactive waste, and other systems that may contain radioactivity shall be designed to ensure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage cooling under accident conditions.	The fuel storage and handling, radioactive waste, and other systems that may contain radioactivity shall be designed to ensure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components ~~important to safety~~ which are safety significant, (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~~ safety significance of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage cooling under accident conditions.
Basis:	The ARDC is non-technology specific and therefore is applicable to the KP-FHR. The basis for the modification to “important to safety” is provided in KP-FHR PDC 1. The SFR-DC and MHTGR-DC are the same as the ARDC.	
Source:	RG 1.232, Appendix A, Criterion 61	

Title:	62: Prevention of criticality in fuel storage and handling.
KP-FHR PDC:	Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.
Position:	The KP-FHR PDC 62 adopts the language from ARDC 62 of RG 1.232, Appendix A with no modifications.
Basis:	The ARDC is non-technology specific and therefore is applicable to the KP-FHR. The language in this ARDC, which adopts the GDC with no changes, is applicable and requires no KP-FHR specific modification. The SFR-DC and MHTGR-DC are the same as the ARDC.
Source:	RG 1.232, Appendix A, Criterion 62

Title:	63: Monitoring fuel and waste storage.
KP-FHR PDC:	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.
Position:	The KP-FHR PDC 63 adopts the language from ARDC 63 of RG 1.232, Appendix A.
Basis:	The ARDC is non-technology specific and therefore is applicable to the KP-FHR. The language in this ARDC, which adopts the GDC with no changes, is applicable and requires no KP-FHR specific modification. The SFR-DC and MHTGR-DC are the same as the ARDC.
Source:	RG 1.232, Appendix A, Criterion 63

Title:	64: Monitoring radioactivity releases.
KP-FHR PDC:	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.
Position:	The KP-FHR PDC 64 adopts the language from MHTGR-DC 64 of RG 1.232, Appendix C.
Basis:	The MHTGR-DC is adopted because the KP-FHR design does not include a containment atmosphere as described in the ARDC and instead relies on the barriers that comprise a functional containment design as described in KP-FHR PDC 16. The SFR-DC is not adopted because it contains references to “sodium” which is not present in KP-FHR.
Source:	RG 1.232, Appendix C, Criterion 64

VII. Additional KP-FHR PDC

<table>
<tr><td>Title:</td><td colspan="2">70: Reactor coolant purity control.</td></tr>
<tr><td>KP-FHR PDC:</td><td colspan="2">Systems shall be provided as necessary to maintain the purity of reactor coolant within specified design limits. These limits shall be based on consideration of (1) chemical attack, (2) fouling and plugging of passages, and (3) radionuclide concentrations, and (4) air or moisture ingress as a result of a leak of cover gas.</td></tr>
<tr><td rowspan="3">Position:</td><td colspan="2">The KP-FHR PDC 70 adopts the language from SFR-DC 71 of RG 1.232, Appendix B with modifications as shown below.</td></tr>
<tr><td>RG 1.232, Appendix B, Criterion 71</td><td>KP-FHR PDC 70</td></tr>
<tr><td>Primary coolant and cover gas purity control.

Systems shall be provided as necessary to maintain the purity of primary coolant sodium and cover gas within specified design limits. These limits shall be based on consideration of (1) chemical attack, (2) fouling and plugging of passages, and (3) radionuclide concentrations, and (4) air or moisture ingress as a result of a leak of cover gas.</td><td>~~Primary~~ Reactor coolant ~~and cover gas~~ purity control.

Systems shall be provided as necessary to maintain the purity of ~~primary~~ reactor coolant ~~sodium and cover gas~~ within specified design limits. These limits shall be based on consideration of (1) chemical attack, (2) fouling and plugging of passages, and (3) radionuclide concentrations, and (4) air or moisture ingress as a result of a leak of cover gas.</td></tr>
<tr><td>Basis:</td><td colspan="2">The SFR-DC 71 is adopted because the KP-FHR design utilizes a molten salt coolant system with a cover gas to support inventory control as does the SFR. The purity of the molten salt coolant is important for the structural integrity of components within the reactor vessel and to ensure that flow paths remain clear and do not impact decay heat removal. In addition, impurities could affect the radioactivity retention properties of the coolant. These are the implied safety functions of maintaining the reactor coolant purity in the SFR and in the KP-FHR. Therefore, maintaining the coolant purity is an appropriate design criteria for the KP-FHR. However, in the SFR, cover gas purity is also independently important to prevent energetic interaction between the cover gas and the primary coolant sodium. Given the different coolants in the KP-FHR, energetic interactions are not thermodynamically favored between Flibe and contaminants of the cover gas. Cover gas purity in the KP-FHR is one of the design options to maintain coolant purity, but is not the only means. Because the energetic interactions of sodium and cover gas contaminant moisture are not present in an FHR, the function of maintaining cover gas purity is not required as a design criteria. Therefore, the referral to cover gas purity is removed. If cover gas purity is required as a design solution to meet the chemistry control requirements of the reactor coolant specified by the PDC, then cover gas purity monitoring will be credited for demonstrating conformance to the PDC.
Although, the phrase “maintain the purity of the reactor coolant” does not specifically mention initial purity, the initial coolant purity design limits are established in consideration of the affects of chemical attack, fouling, radionuclide accumulation, or air or moisture ingress.</td></tr>
<tr><td>Source:</td><td colspan="2">RG 1.232, Appendix B, Criterion 71</td></tr>
</table>

Title:	71: Reactor coolant heating systems.
KP-FHR PDC:	Heating systems shall be provided for systems and components that are safety significant, and that contain or could be required to contain reactor coolant in liquid form. These heating systems and their controls shall be appropriately designed to ensure that the temperature distribution and rate of change of temperature in systems and components containing reactor coolant are maintained within design limits assuming a single failure. If plugging of any cover gas line due to condensation or plate out of reactor coolant aerosol or vapor could prevent accomplishing a safety function, the temperature control and the relevant corrective measures associated with that line shall be considered safety significant.
Position:	The KP-FHR PDC 71 adopts the language from SFR-DC 72 of RG 1.232, Appendix B with modifications as shown below.

RG 1.232, Appendix B, Criterion 72	KP-FHR PDC 71
Sodium heating systems Heating systems shall be provided for systems and components that are important to safety, and that contain or could be required to contain sodium. These heating systems and their controls shall be appropriately designed to ensure that the temperature distribution and rate of change of temperature in systems and components containing sodium are maintained within design limits assuming a single failure. If plugging of any cover gas line due to condensation or plate out of sodium aerosol or vapor could prevent accomplishing a safety function, the temperature control and the relevant corrective measures associated with that line shall be considered important to safety.	*~~Sodium~~ Reactor coolant heating systems* Heating systems shall be provided for systems and components that are ~~important to safety~~safety significant, and that contain or could be required to contain ~~sodium~~ reactor coolant in liquid form. These heating systems and their controls shall be appropriately designed to ensure that the temperature distribution and rate of change of temperature in systems and components containing ~~sodium~~ reactor coolant are maintained within design limits assuming a single failure. If plugging of any cover gas line due to condensation or plate out of ~~sodium~~ reactor coolant aerosol or vapor could prevent accomplishing a safety function, the temperature control and the relevant corrective measures associated with that line shall be considered ~~important to safety~~safety significant.

Basis:	The SFR-DC 72 is adopted because the KP-FHR design utilizes a molten salt coolant system to remove heat from the core and also uses heating systems to ensure that the coolant remains within design limits. This is similar to the SFR design. One modification to the PDC wording is made to remove reference to sodium because KP-FHR uses Flibe as the reactor coolant instead of sodium. The basis for the modification to "important to safety" is provided in KP-FHR PDC 1.
Source:	RG 1.232, Appendix B, Criterion 72

<table>
<tr><td>Title:</td><td colspan="2">73: Reactor coolant system interfaces.</td></tr>
<tr><td>KP-FHR PDC:</td><td colspan="2">When the reactor coolant system interfaces with a structure, system, or component containing fluid that is chemically incompatible with the primary coolant, the interface location shall be designed to ensure that the primary coolant is separated from the chemically incompatible fluid by two redundant, passive barriers. When the reactor coolant system interfaces with a structure, system, or component containing fluid that is chemically compatible with the reactor coolant, then the interface location may be a single passive barrier provided that the postulated leakage at the interface location does not result in failure of the intended safety functions of structures, systems or components which are safety significant or result in exceeding the specified acceptable system radionuclide release design limits during normal operation, anticipated operational occurrences, shutdown, and accident conditions.</td></tr>
<tr><td rowspan="3">Position:</td><td colspan="2">The KP-FHR PDC 73 adopts the language from SFR-DC 78 of RG 1.232, Appendix B with modifications as shown below.</td></tr>
<tr><td>RG 1.232, Appendix B, Criterion 78</td><td>KP-FHR PDC 73</td></tr>
<tr><td>Primary coolant system interfaces

When the primary coolant system interfaces with a structure, system, or component containing fluid that is chemically incompatible with the primary coolant, the interface location shall be designed to ensure that the primary coolant is separated from the chemically incompatible fluid by two redundant, passive barriers. When the primary coolant system interfaces with a structure, system, or component containing fluid that is chemically compatible with the primary coolant, then the interface location may be a single passive barrier provided that the following conditions are met:

(1) postulated leakage at the interface location does not result in failure of the intended safety functions of structures, systems or components important to safety or result in exceeding the fuel design limits

(2) the fluid contained in the structure, system, or component is maintained at a higher pressure than the primary coolant during normal operation, anticipated operational occurrences, shutdown, and accident conditions.</td><td>~~Primary~~ <u>Reactor</u> coolant system interfaces

When the reactor coolant system interfaces with a structure, system, or component containing fluid that is chemically incompatible with the primary coolant, the interface location shall be designed to ensure that the primary coolant is separated from the chemically incompatible fluid by two redundant, passive barriers. When the ~~primary~~ <u>reactor</u> coolant system interfaces with a structure, system, or component containing fluid that is chemically compatible with the ~~primary~~ <u>reactor</u> coolant, then the interface location may be a single passive barrier provided that the ~~following conditions are met: (1)~~ postulated leakage at the interface location does not result in failure of the intended safety functions of structures, systems or components ~~important to safety~~ <u>which are safety significant</u> or result in exceeding the ~~fuel design limits~~ <u>specified acceptable system radionuclide release design limits</u> ~~(2) the fluid contained in the structure, system, or component is maintained at a higher pressure than the primary coolant~~ during normal operation, anticipated operational occurrences, shutdown, and accident conditions.</td></tr>
</table>

Basis:	The KP-FHR includes an intermediate heat transport system between the reactor coolant and the steam system. Therefore, this SFR-DC is evaluated and is considered applicable, in part, to the KP-FHR. However, there are significant differences between the coolants utilized in FHRs and SFRs and modification to the SFR-DC is warranted. The KP-FHR primary and intermediate coolants are compatible, where “compatible” is defined as not producing exothermic reactions upon mixing, such as the reaction from mixing sodium and water. Conditional sentence 1 is changed to replace “fuel design limits” with “specified acceptable system radionuclide release design limits” (SARRDL) to be consistent with the KP-FHR figure of merit as described in the basis for PDC 10. Conditional sentence 2 is deleted because the consequences of reactor coolant leakage in an FHR are more similar to steam generator tube leakage in a PWR (reactor coolant to intermediate) than intermediate heat exchanger leakage in an SFR (intermediate to reactor coolant) and as such, the direction of leakage does not possess the same potential consequences as in SFRs. The reactor coolant activity will be regularly monitored and kept below technical specification-controlled levels. Leakage from the reactor coolant to the intermediate coolant is monitored consistent with KP-FHR PDC 60. The basis for the replacement of “important to safety” with “safety-significant” is described in KP-FHR PDC 1.
Source:	RG 1.232, Appendix B, Criterion 78

Title:	74: Reactor vessel and reactor system structural design basis.
KP-FHR PDC:	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.
Position:	The KP-FHR PDC 74 adopts the language for MHTGR-DC 70 in RG 1.232, Appendix C with no modifications.
Basis:	The KP-FHR includes a passive heat removal system; therefore, this MHTGR-DC is evaluated for applicability. Similar to the MHTGR, the KP-FHR relies on the reactor vessel structures to permit the successful performance of the passive residual heat removal system and permit sufficient insertion of the neutron absorbers to provide for reactor shutdown. Therefore, this MHTGR-DC is appropriate for inclusion in the KP-FHR PDC.
Source:	RG 1.232, Appendix C, Criterion 70

<table>
<tr><td>Title:</td><td colspan="2">75: Reactor building design basis.</td></tr>
<tr><td>KP-FHR PDC:</td><td colspan="2">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.</td></tr>
<tr><td rowspan="3">Position:</td><td colspan="2">The KP-FHR PDC adopt the language from MHTGR-DC 71 with modifications as shown below.</td></tr>
<tr><td>RG 1.232, Appendix C, Criterion 71</td><td>KP-FHR PDC 75</td></tr>
<tr><td>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.</td><td>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~~.</td></tr>
<tr><td>Basis:</td><td colspan="2">The KP-FHR employs a reactor building to provide protection from external events for the reactor core and safety significant structures, systems and components. Therefore, this MHTGR-DC is evaluated for applicability.
In RG 1.232, the basis for functional containment (MHTGR-DC 16) establishes an expectation that criterion for structural protection associated with the functional containment approach be included. As the KP-FHR also incorporates functional containment design philosophy, the KP-FHR PDC also include this additional DC, with some modification.
The reactor building envisioned for the KP-FHR is expected to be significantly smaller and lower profile than the traditional buildings utilized in MHTGR design because of the physically smaller dimensions of the KP-FHR reactor vessel. The reactor building as defined here is the physical structure which allows for the protection of the geometry for passive removal of residual heat from the reactor core, and is not expected to encompass other physical structures outside the reactor building.
Reference to depressurization accidents in the original criterion specific to MHTGR has been removed due to the low operating pressure of the KP-FHR.</td></tr>
<tr><td>Source:</td><td colspan="2">RG 1.232, Appendix C, Criterion 71</td></tr>
</table>

<table>
<tr><td>Title:</td><td colspan="2">76: Provisions for periodic reactor building inspection.</td></tr>
<tr><td>KP-FHR PDC:</td><td colspan="2">The reactor building shall be designed to permit (1) appropriate periodic inspection of all important structural areas, and (2) an appropriate surveillance program.</td></tr>
<tr><td rowspan="3">Position:</td><td colspan="2">The KP-FHR PDC adopt MHTGR-DC 72 with the modification as shown below.</td></tr>
<tr><td>RG 1.232, Appendix C, Criterion 72</td><td>KP-FHR PDC</td></tr>
<tr><td>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.</td><td>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.</td></tr>
<tr><td>Basis:</td><td colspan="2">The basis for inclusion of this MHTGR-DC is discussed in KP-FHR PDC 75.</td></tr>
<tr><td>Source:</td><td colspan="2">RG 1.232, Appendix C, Criterion 72</td></tr>
</table>

APPENDIX B. RG 1.232 DESIGN CRITERIA NOT APPLICABLE TO THE KP-FHR

I. Overall Requirements

None.

II. Multiple Barriers

None.

III. Reactivity Control

Title:	27: Combined reactivity control systems capability.
KP-FHR PDC:	Deleted per RG 1.232, Appendix A
Position:	The KP-FHR PDC adopts the changes in RG 1.232 for the ARDC with no KP-FHR specific modification.
Basis:	This PDC is deleted because it has been combined with PDC 26 as discussed in RG 1.232, Appendix A.
Source:	RG 1.232, Appendix A, Criterion 27

IV. Fluid Systems/Heat Transport

Title:	38: Containment heat removal.
KP-FHR PDC:	Not Applicable
Position:	The KP-FHR PDC 38 adopts the language from MHTGR-DC 38 of RG 1.232, Appendix C which states that this design criteria is not applicable.
Basis:	The KP-FHR design does not include a pressure-retaining containment structure. Therefore, this ARDC is not applicable to the KP-FHR. The KP-FHR design relies on a functional containment approach as described in KP-FHR PDC 16.
Source:	RG 1.232, Appendix C, Criterion 38

Title:	39: Inspection of containment heat removal system.
KP-FHR PDC:	Not Applicable
Position:	The KP-FHR PDC 39 adopts the language from MHTGR-DC 39 of RG 1.232, Appendix C which states that this design criteria is not applicable.
Basis:	The KP-FHR design does not include a pressure-retaining containment structure. Therefore, this ARDC is not applicable to the KP-FHR. The KP-FHR design relies on a functional containment approach as described in KP-FHR PDC 16.
Source:	RG 1.232, Appendix C, Criterion 39

Title:	40: Testing of containment heat removal system.
KP-FHR PDC:	Not Applicable
Position:	The KP-FHR PDC 40 adopts the language from MHTGR-DC 40 of RG 1.232, Appendix C which states that this design criteria is not applicable.
Basis:	The KP-FHR design does not include a pressure-retaining containment structure. Therefore, ARDC 16 is not applicable to the KP-FHR. The KP-FHR design relies on a functional containment approach as described in KP-FHR PDC 16.
Source:	RG 1.232, Appendix C, Criterion 40

Title:	41: Containment atmosphere cleanup.
KP-FHR PDC:	Not Applicable
Position:	The KP-FHR PDC 41 adopts the language from MHTGR-DC 41 of RG 1.232, Appendix C which states that this design criteria is not applicable.
Basis:	The KP-FHR design does not include a pressure-retaining containment structure. Therefore, ARDC 16 is not applicable to the KP-FHR. The KP-FHR design relies on a functional containment approach as described in KP-FHR PDC 16.
Source:	RG 1.232, Appendix C, Criterion 41

Title:	42: Inspection of containment atmosphere cleanup systems.
KP-FHR PDC:	Not Applicable
Position:	The KP-FHR PDC 42 adopts the language from MHTGR-DC 42 of RG 1.232, Appendix C which states that this design criteria is not applicable.
Basis:	The KP-FHR design does not include a pressure-retaining containment structure. Therefore, ARDC 16 is not applicable to the KP-FHR. The KP-FHR design relies on a functional containment approach as described in KP-FHR PDC 16.
Source:	RG 1.232, Appendix C, Criterion 42

Title:	43: Testing of containment atmosphere cleanup systems.
KP-FHR PDC:	Not Applicable
Position:	The KP-FHR PDC 43 adopts the language from MHTGR-DC 43 of RG 1.232, Appendix C which states that this design criteria is not applicable.
Basis:	The KP-FHR design does not include a pressure-retaining containment structure. Therefore, ARDC 16 is not applicable to the KP-FHR. The KP-FHR design relies on a functional containment approach as described in KP-FHR PDC 16.
Source:	RG 1.232, Appendix C, Criterion 43

V. Reactor Containment

Title:	50: Containment design basis.
KP-FHR PDC:	Not Applicable
Position:	The KP-FHR PDC 50 adopts the language from MHTGR-DC 50 of RG 1.232, Appendix C which states that this design criteria is not applicable.
Basis:	The KP-FHR design does not include a pressure-retaining containment structure. Therefore, ARDC 50 is not applicable to the KP-FHR. The KP-FHR design relies on a functional containment approach as described in KP-FHR PDC 16.
Source:	RG 1.232, Appendix C, Criterion 50

Title:	51: Fracture prevention of containment pressure boundary.
KP-FHR PDC:	Not Applicable
Position:	The KP-FHR PDC 51 adopts the language from MHTGR-DC 51 of RG 1.232, Appendix C which states that this design criteria is not applicable.
Basis:	The KP-FHR design does not include a pressure-retaining containment structure. Therefore, ARDC 51 is not applicable to the KP-FHR. The KP-FHR design relies on a functional containment approach as described in KP-FHR PDC 16.
Source:	RG 1.232, Appendix C, Criterion 51

Title:	52: Capability for containment leakage rate testing.
KP-FHR PDC:	Not Applicable
Position:	The KP-FHR PDC 52 adopts the language from MHTGR-DC 52 of RG 1.232, Appendix C which states that this design criteria is not applicable.
Basis:	The KP-FHR design does not include a pressure-retaining containment structure. Therefore, ARDC 52 is not applicable to the KP-FHR. The KP-FHR design relies on a functional containment approach as described in KP-FHR PDC 16.
Source:	RG 1.232, Appendix C, Criterion 52

Title:	53: Provisions for containment testing and inspection.
KP-FHR PDC:	Not Applicable
Position:	The KP-FHR PDC 53 adopts the language from MHTGR-DC 53 of RG 1.232, Appendix C which states that this design criteria is not applicable.
Basis:	The KP-FHR design does not include a pressure-retaining containment structure. Therefore, ARDC 53 is not applicable to the KP-FHR. The KP-FHR design relies on a functional containment approach as described in KP-FHR PDC 16.
Source:	RG 1.232, Appendix C, Criterion 53

Title:	54: Piping systems penetrating containment.
KP-FHR PDC:	Not Applicable
Position:	The KP-FHR PDC 54 adopts the language from MHTGR-DC 54 of RG 1.232, Appendix C which states that this design criteria is not applicable.
Basis:	The KP-FHR design does not include a pressure-retaining containment structure. Therefore, ARDC 54 is not applicable to the KP-FHR. The KP-FHR design relies on a functional containment approach as described in KP-FHR PDC 16.
Source:	RG 1.232, Appendix C, Criterion 54

Title:	55: Reactor coolant pressure boundary penetrating containment.
KP-FHR PDC:	Not Applicable
Position:	The KP-FHR PDC 55 adopts the language from MHTGR-DC 55 of RG 1.232, Appendix C which states that this design criteria is not applicable.
Basis:	The KP-FHR design does not include a pressure-retaining containment structure. Therefore, ARDC 55 is not applicable to the KP-FHR. The KP-FHR design relies on a functional containment approach as described in KP-FHR PDC 16.
Source:	RG 1.232, Appendix C, Criterion 55

Title:	56: Primary containment isolation.
KP-FHR PDC:	Not Applicable
Position:	The KP-FHR PDC 56 adopts the language from MHTGR-DC 56 of RG 1.232, Appendix C which states that this design criteria is not applicable.
Basis:	The KP-FHR design does not include a pressure-retaining containment structure. Therefore, ARDC 56 is not applicable to the KP-FHR. The KP-FHR design relies on a functional containment approach as described in KP-FHR PDC 16.
Source:	RG 1.232, Appendix C, Criterion 56

Title:	57: Closed system isolation valves.
KP-FHR PDC:	Not Applicable
Position:	The KP-FHR PDC 57 adopts the language from MHTGR-DC 57 of RG 1.232, Appendix C which states that this design criteria is not applicable.
Basis:	The KP-FHR design does not include a pressure-retaining containment structure. Therefore, ARDC 57 is not applicable to the KP-FHR. The KP-FHR design relies on a functional containment approach as described in KP-FHR PDC 16.
Source:	RG 1.232, Appendix C, Criterion 57

VI. Fuel and Radioactivity Control

None.

VII. Additional SFR-DC

Title:	SFR-DC 70: Intermediate coolant system.
KP-FHR PDC:	Not Applicable
Position:	SFR-DC 70 does not apply to the KP-FHR and the language is not adopted.
Basis:	This SFR-DC 70 is not applicable to the KP-FHR. The NRC rationale in RG 1.232 and NUREG-1368 (Reference 4) suggests the need for a separate criterion for the intermediate coolant system (ICS) for SFRs. However, the same basis does not apply for the KP-FHR. The need for separate criteria is based on assumed functions for the intermediate loop as well as an assumed use of sodium coolant, consistent with an SFR design. The NRC rationale is summarized as follows: (1) to ensure that the ICS does not impact the safety of the primary coolant system, (2) to ensure that radioactivity in the primary coolant system does not transfer into the power conversion system, and (3) to ensure that the ICS is designed to minimize the possibility of a large, uncontrolled release of sodium. The intent of items 2 and 3 above is interpreted to be based on a need for maintaining the integrity of the intermediate coolant boundary to preclude a large release of radioactivity through the intermediate loop which could result from an adverse chemical reaction with the sodium coolant. For the KP-FHR, neither the primary coolant loop nor intermediate loop are sodium based and this consideration is not applicable. With respect to item 1, protection from adverse interactions of the primary coolant loop are already addressed by KP-FHR PDC 4 and PDC 30 and a separate criterion is not necessary under the stated context. Furthermore, there is no additional analogous requirement for LWRs for the design of the secondary system to protect the reactor coolant boundary because such protection is covered under GDCs 4 and 30. Additionally, the requirement to maintain the integrity of the primary coolant boundary is addressed in RG 1.232, Appendix C, Criterion 70 for MHTGR which is adopted for the KP-FHR as PDC 74. KP-FHR PDC 73 addresses fluid compatibility of the intermediate system with the primary system. Therefore, inclusion of SFR DC 70 is not warranted for the KP-FHR.
Source:	RG 1.232, Appendix B, Criterion 70

Title:	SFR-DC 73: Sodium leakage detection and reaction prevention and mitigation.
KP-FHR PDC:	Not Applicable
Position:	The SFR-DC 73 does not apply to the KP-FHR and the language is not adopted.
Basis:	SFR-DC 73 is not adopted for the KP-FHR. While the NRC rationale for including this criteria, as indicated in RG 1.232, Appendix B, is based on precluding an adverse chemical reaction between sodium and air and sodium and concrete. For the KP-FHR, neither the reactor coolant loop nor intermediate loop are sodium based and this consideration is not applicable.
Source:	RG 1.232, Appendix B, Criterion 73

Title:	SFR-DC 74: Sodium/water reaction prevention/mitigation.
KP-FHR PDC:	Not Applicable
Position:	The SFR-DC 74 does not apply to the KP-FHR and the language is not adopted.
Basis:	SFR-DC 74 is not adopted for the KP-FHR. While the KP-FHR does include an intermediate loop, the NRC rationale for including this criteria, as indicated in RG 1.232, Appendix B, is based precluding an adverse chemical reaction between sodium and water coolants. For the KP-FHR, neither the reactor coolant loop nor intermediate loop are sodium based and this consideration is not applicable.
Source:	RG 1.232, Appendix B, Criterion 74

Title:	SFR-DC 75: Quality of intermediate coolant boundary.
KP-FHR PDC:	Not Applicable
Position:	The SFR-DC 75 does not apply to the KP-FHR and is not adopted.
Basis:	SFR-DC 75 is not adopted for the KP-FHR. While the KP-FHR does include an intermediate loop, the NRC rationale for including this criteria, as indicated in RG 1.232, Appendix B, is based on precluding an adverse chemical reaction between sodium and water coolants. For the KP-FHR, neither the reactor coolant loop nor intermediate loop are sodium based and this consideration is not applicable. As discussed in the basis for SFR DC 70, protection of the primary coolant boundary from adverse conditions or events in the intermediate coolant loop are provided by existing PDCs 4, 30, 73, and 74.
Source:	RG 1.232, Appendix B, Criterion 75

Title:	SFR-DC 76: Fracture prevention of the intermediate coolant boundary.
KP-FHR PDC:	Not Applicable
Position:	The SFR-DC 76 does not apply to the KP-FHR and is not adopted.
Basis:	SFR-DC 76 is not adopted for the KP-FHR. While the KP-FHR does include an intermediate loop, the NRC rationale for including this criteria, as indicated in RG 1.232, Appendix B, is based precluding an adverse chemical reaction between sodium and water coolants. For the KP-FHR, neither the reactor coolant loop nor intermediate loop are sodium based and this consideration is not applicable. As discussed in the basis for SFR DC 70, protection of the primary coolant boundary from adverse conditions or events in the intermediate coolant loop are provided by existing PDCs 4, 30, 73, and 74.
Source:	RG 1.232, Appendix B, Criterion 76

Title:	SFR-DC 77: Inspection of intermediate coolant boundary.
KP-FHR PDC:	Not Applicable
Position:	The SFR-DC 77 does not apply to the KP-FHR and is not adopted.
Basis:	SFR-DC 77 is not adopted for the KP-FHR. While the KP-FHR does include an intermediate loop, the KP-FHR intermediate loop physically separates the reactor coolant boundary from the conventional island. The intermediate loop coolant is chemically inert compared to sodium (which is the basis for the SFR-DC) and does not exhibit energetic interactions with water or the reactor coolant. Consistent with the basis in RG 1.232, Appendix B for this SFR-DC, the KP-FHR intermediate loop does not perform safety significant functions that would necessitate a leaktight boundary. The intermediate heat transport system is designed such that leakage from the primary to the intermediate loop and associated diffusion of radionuclides is limited to acceptable limits consistent with KP-FHR PDC 60. Additionally, applicable American Society of Mechanical Engineers codes govern inspection, testing and quality control for the intermediate coolant boundary components. As discussed in the basis for SFR DC 70, protection of the primary coolant boundary from adverse conditions or events in the intermediate coolant loop are provided by existing PDCs 4, 30, 73, and 74.
Source:	RG 1.232, Appendix B, Criterion 77

Title:	SFR-DC 79: Cover gas inventory maintenance.
KP-FHR PDC:	Not Applicable
Position:	The SFR-DC 79 does not apply to the KP-FHR and is not adopted.
Basis:	SFR-DC 79 is not adopted for the KP-FHR. The SFR-DCs state that the cover gas in an SFR “performs an important to safety function by protecting the sodium coolant from chemical reactions.” Due to the different chemistry of the reactor coolant from sodium, energetic coolant-air interactions are not thermodynamically favored in the KP-FHR. As such, maintaining the cover gas inventory is not a safety function and so the SFR-DC is not applicable.
Source:	RG 1.232, Appendix B, Criterion 79