There is no proprietary material in this Safety Evaluation.

Safety Evaluation

"Principal Design Criteria for the Kairos Power Fluoride Salt-Cooled,

High Temperature Reactor"

Docket No. 99902069

## 1.0 INTRODUCTION

By letter dated December 21, 2018, Kairos Power LLC (Kairos, 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 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, Kairos 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," 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 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, Kairos 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. 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." Kairos 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 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, "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." 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, the reactor coolant, and the TRISO fuel 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 the 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 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 report 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,” and NEI 18-04, “Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development,” 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.

<u>PDC with No Changes</u>

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.

<u>PDC with Single Change - Using Safety Significant Instead of Important to Safety</u>

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, 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 pressure 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 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 72, 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 Kairos 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 72, to be acceptable because a PDC that requires maintaining the salt as a liquid 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 72 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 Kairos 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.

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. 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. 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. 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. 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. 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. SFR-DC 79 states that the maintenance of cover gas inventory is needed to ensure primary coolant sodium design limits are not exceeded. NUREG-1368, 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. The proposed PDCs are sufficiently broad to provide an appropriate set of requirements for the KP-FHR.

## 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 reactor fuel with reactor 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 staff's approval of the applicant's implementation of the guidance in DG-1353 and NEI 18-04.

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 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.

## 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, Acceptance for Review of Kairos Power LLC, “Principal Design Criteria for the Kairos Power Fluoride Salt-Cooled, High Temperature Reactor Topical Report,” ADAMS Accession No. ML19059A355.

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

Principal Contributor: Boyce Travis, NRR

Date: March 4th, 2020.