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KAIROS POWER LLC – FINAL SAFETY EVALUATION OF TOPICAL REPORT KP-TR-020-P, "SAFETY ANALYSIS METHODOLOGY FOR THE KAIROS POWER FLUORIDE SALT-COOLED HIGH-TEMPERATURE TEST REACTOR," REVISION 1 (EPID L-2024-TOP-0022)

SPONSOR AND SUBMITTAL INFORMATION

Sponsor: Kairos Power LLC (Kairos)

Sponsor Address: 707 W. Tower Ave, Suite A
Alameda, CA 94501

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Brief Description of the Topical Report: On June 4, 2024, Kairos Power LLC (Kairos) submitted topical report (TR) KP-TR-020, "Safety Analysis Methodology for the Kairos Power Fluoride Salt-Cooled High-Temperature Test Reactor," Revision 0, for the U.S. Nuclear Regulatory Commission (NRC) staff (the staff) review. On July 29, 2025, Kairos submitted Revision 1 of the TR (ML25210A576). The TR documents a safety analysis methodology for evaluating the maximum hypothetical accident (MHA) and postulated events for the Kairos Power Fluoride Salt-Cooled, High Temperature Reactor (KP-FHR) test reactor. The purpose of the safety analysis methodology is to demonstrate that the dose consequences of the postulated events are bounded by an MHA that has acceptable dose consequences. The TR identifies the computer codes used and discusses related verification and validation efforts for the postulated event analyses. The TR describes the base input model for the KP-FHR test reactor and the event-specific biases and sensitivity studies that are designed to address the uncertainties and identify the limiting postulated events.

REGULATORY EVALUATION

Regulatory Basis

Title 10 of the *Code of Federal Regulations* (10 CFR) 50.34(b) requires that each application for an operating license (OL) include a final safety analysis report (FSAR) that provides a description and safety assessment of the plant design features intended to mitigate the radiological consequences of accidents. The FSAR must demonstrate compliance with the radiological consequence evaluation factors for offsite doses at the exclusion area boundary

(EAB) and outer boundary of the low population zone (LPZ) as required by 10 CFR Part 100, “Reactor Site Criteria.”

Furthermore, 10 CFR 50.34(b)(4) requires, in part, analysis and evaluation of the design and performance of structures, systems, and components (SSCs) of the facility to assess the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of SSCs provided for the prevention of accidents and the mitigation of the consequences of accidents.

The safety analysis methodology presented in the TR is used to evaluate the postulated event dose consequences for the KP-FHR test reactor. The safety analysis methodology follows the guidance in NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors,” (ML042430055).

Consistent with the NUREG-1537 guidance, the safety analysis methodology presented in the TR comprises two major analyses: MHA and postulated event analyses. The MHA is utilized to demonstrate that the radiological consequences from a bounding postulated event result in projected dose levels within the regulatory dose criteria.

The MHA described in TR section 2 is based on the approved KP-FHR mechanistic source term (MST) methodology described in TR KP-TR-012-NP-A, “KP-FHR Mechanistic Source Term Methodology” (ML22136A291). However, Section 2 of the TR presents deviations from the approved MST methodology as well as from the MHA analysis approved in the Hermes and Hermes 2 construction permit application reviews. The staff used guidance in NUREG-1537 to evaluate the updates to the MHA.

The postulated event analysis evaluation model (EM) described in the TR is based on the Kairos Power systems analysis code (KP-SAM). The KP-SAM EM is developed in accordance with the applicable portions of Regulatory Guide (RG) 1.203, “Transient and Accident Analysis Methods” (ML053500170). Although NUREG-1537 does not explicitly require non-power reactor applicants to follow RG 1.203 for safety analysis methodology development, the principles and expectations outlined in RG 1.203 are consistent with NUREG-1537’s guidance. Specifically, NUREG-1537 requires that safety analysis models address key elements such as assumptions, approximations, validation, and uncertainty, which are also central to RG 1.203. Accordingly, the use of RG 1.203 guidance for developing and assessing the KP-SAM EM for analysis of transient and accident conditions in KP-FHR is appropriate. In particular, RG 1.203 outlines a structured 20-step process known as the Evaluation Model Development and Assessment Process (EMDAP), organized into 4 elements. Accordingly, the staff’s review of the KP-SAM postulated event EM follows the applicable EMDAP guidance.

NUREG-1537, part 1, chapter 13, also describes the following objectives for the postulated event analysis:

- Ensure that enough events have been considered to include any accident with significant radiological consequences. Rejection of a potential event should be justified in the discussions.
- Categorize the initiating events and scenarios by type and likelihood of occurrence so that only the limiting cases in each group must be quantitatively analyzed.
- Develop and apply consistent, specific acceptance criteria for the consequences of each postulated event.

The staff considered these objectives for the evaluation of postulated event analysis presented in the TR.

Section 1.2 of the TR states that the safety analysis methodology is used to conform to KP-FHR principal design criteria (PDC) 19, which provides control room design criteria, including a radiological habitability accident dose criterion for control room personnel. The PDC for the KP-FHR design were reviewed and approved by the staff in TR KP-TR-003-NP-A, “Principal Design Criteria for the Kairos Power Fluoride Salt-Cooled, High Temperature Reactor,” (ML20167A174).

The staff considered the regulations, guidance, and applicable previously approved methodologies and KP-FHR PDC 19 in its review of the TR. However, evaluation of safety analysis methodology implementation will occur as part of the OL review.

TECHNICAL EVALUATION

As described in section 1 of the TR, Kairos is requesting the staff’s approval of the safety analysis methodology as an appropriate way to determine the acceptability of postulated event dose consequences by comparing them to an MHA with acceptable dose consequences. Kairos intends to use the safety analysis methodology in a licensing application for a KP-FHR test reactor to address the requirements of 10 CFR 100.11, “Determination of exclusion area, low population zone, and population center distance,” and conformance with KP-FHR PDC 19. Section 1 of this SE summarizes the major components of the TR and clarifies the staffs evaluation of each major component.

1. Topical Report Overview

The TR consists of the following major sections and appendix:

Section 1 of the TR provides a brief description of KP-FHR technology and identifies key design features (i.e., SSCs) modeled by the safety analysis methodology. Limitation 5 in TR section 6.2 states that the safety analysis methodology is based on KP-FHR design features provided in section 1.1, and deviations from these design features will be justified in future applications. The staff includes this limitation through **Limitation and Condition 1** in the “Limitations and Conditions” section of this SE. The staff considered the design features described in section 1.1 of the TR throughout its technical evaluation of this TR. TR section 1 also identifies specific technical areas for which Kairos is requesting staff review and approval in this TR.

Section 2 of the TR describes the analysis approach to calculate the dose consequences of the MHA with refinements from previously approved methods and analysis (i.e., KP-TR-012-NP-A, Hermes 1 and 2 construction permit applications). This section also includes information to support disposition of the MST methodology TR safety evaluation (SE) limitations and conditions 2 and 5. The MHA in section 2 is also used to demonstrate conformance to the control room dose criterion in KP-FHR PDC 19. The staff’s evaluation of the updated MHA and its adequacy to address KP-FHR PDC 19 is in section 2 of this SE.

Section 3 of the TR addresses Element 1 of EMDAP (Establish Requirements for Evaluation Model Capability) by describing the postulated event categories addressed by the safety analysis methodology and the figures of merit (FOMs) selected for the safety analysis methodology. TR section 3 also summarizes how SSCs and associated phenomena identified through the phenomena identification and ranking table (PIRT) process are addressed in the safety analysis methodology. The staff evaluated the adequacy of proposed postulated event categories and FOMs and reviewed the safety analysis methodology approach for addressing the PIRT phenomena in SE section 3.

Section 4 of the TR addresses, in part, Element 3 of EMDAP by describing the KP-SAM-based EM in sections 4.1 and 4.3. Section 4.2 addresses, in part, Element 2 (Develop Assessment Base) and Element 4 (Assess Evaluation Model Adequacy) of EMDAP. The staff's review of the KP-SAM code models and correlations, base input model, and proposed validations is in SE section 4.

Section 5 of the TR addresses, in part, Element 3 of EMDAP by describing event-specific biases and sensitivity studies for each postulated event category. The staff's evaluation of the proposed event-specific biases and sensitivity studies and its adequacy to account for the safety analysis methodology uncertainties is in SE section 5.

Appendix A of the TR illustrates a generic system response for selected example transients. The sample calculations in appendix A do not represent final design information and Kairos does not request NRC approval for these sample calculations. The staff considered the information in appendix A but does not make any determinations on the sample calculations in appendix A.

2. Maximum Hypothetical Accident

TR section 2 describes the MHA for the KP-FHR test reactor as "a hypothetical set of conditions that postulated a conservative release of radionuclides that bounds a potential release from other postulated events." TR section 2.1 states that, the MHA analysis applies assumed temperature histories to the radioactive material at risk for release (MAR) in the primary system to drive radionuclides out of the system through diffusion and evaporation.

Except for MST methodology and MHA analysis refinements evaluated below, the MHA described in the TR is fundamentally the same as was described in the preliminary safety analysis reports (PSARs) for the Hermes and Hermes 2 facilities (ML23151A745 and ML24144A092) and found acceptable by the staff in sections 13.1.1 and 13.2.1 of their respective SEs to support the issuance of the construction permits (ML23158A268 and ML24200A115). Specifically, the TR states, the MHA is a non-physical scenario with the MHA temperature conditions represented in TR figure 2-1. The temperature profile provided in TR figure 2-1 is identical to the MHA temperature profile used in the Hermes and Hermes 2 PSAR analyses. The modeling of radionuclide release and transport to develop the MHA source term is based primarily on the approved MST methodology. In the MST methodology, the development of event-specific radiological releases to the environment is accomplished by modeling the facility as a set of MAR sources, identifying the succession of barriers to release, and applying a release fraction for each barrier that contains the MAR.

TR section 2 describes MST methodology and MHA analysis refinements from the previously approved MST methodology and the MHA analysis as described in the Hermes and Hermes 2 PSARs in the following areas:

- Argon activation and release models (see SE section 2.1 and 2.3);
- Flibe radionuclide grouping structure (see SE section 2.2);

- Justification of the representative element vapor pressure correlations to address a limitation in the MST methodology TR (see SE section 2.2);
- Isotopic screening criteria (see SE section 2.4); and
- Added methods for evaluation of control room dose consequences (see SE section 2.5).

The staff's review of these refinements is presented below in the order that the MHA information is provided in TR section 2. The remainder of the information in TR section 2 describing the MHA scenario and source term methods, including information on functional containment, is unchanged from the description in the Hermes and Hermes 2 PSARs and is also consistent with the methods in the approved MST methodology. Therefore, these items, which are unchanged, are acceptable as described in the SEs for the Hermes and Hermes 2 construction permits.

2.1 Quantification of MAR

The information in TR section 2.2.1 is the same as described in the Hermes and Hermes 2 PSAR sections 13.1.1 and 13.2.1, except for the description of methods for modeling argon activation and release in TR section 2.2.1.1. Specifically, the TR adds methods for modeling argon-41 (Ar-41), which is generated from the activation of argon-40 (Ar-40) dissolved in Flibe or entrained in bubbles in Flibe.

TR section 2.2.1.1 describes the generation of Ar-41 as the product of neutron activation of argon-40 (Ar-40) in more locations than was previously modeled in the Hermes and Hermes 2 PSARs. In addition to activation of argon in the cover gas and within pores in reflector graphite and pebble carbon matrix, the TR models activation of argon dissolved in Flibe and contained in the bubbles entrained in the Flibe. This change in the Ar-41 generation model was made to support more detailed radionuclide transport modeling in the MHA analysis. Based on its review, the staff finds that the methods described in the TR to account for activation of argon exposed to a neutron flux from the core are acceptable because they use established neutron activation models based on widely used and accepted nuclear cross-section data.

The TR states in the modeling of Ar-41 release from the graphite and pebble carbon matrix that all pores are conservatively assumed to be open to the surface interface with the Flibe. This is a difference from the modeling in the Hermes and Hermes 2 PSAR, which allows for more Ar-41 release from the graphite and pebbles to the Flibe. The Ar-41 solubility-limited pore release model in the TR determines a steady state equilibrium activity distribution in the pores and Flibe. The model uses the tritium mass transfer correlations from the MST methodology to bound the mass transfer rate of argon and uses Henry's Law with the solubility of argon in Flibe. The staff confirmed the basis for the argon release model in audit discussions (ML25197A042), including data on argon solubility in Flibe. Based on its review, the staff finds that the Ar-41 model is acceptable because appropriate argon solubility data is used and the model includes conservatively biased assumptions.

There are no changes from the Hermes and Hermes 2 PSARs descriptions of the modeling of radionuclide transport in fuel (TR section 2.2.2) and transport of MAR from the Flibe to the gas space through bubble burst (TR section 2.2.3.1).

2.2 MAR Release from Flibe

TR section 2.2.3.2 provides a description of the methods to model evaporative release from Flibe. This section includes a modification of the radionuclide grouping structure for radionuclide transport in Flibe as compared to the grouping used in the approved MST methodology. The revised radionuclide groups are listed in TR table 2-1. As stated in the TR, the modified grouping structure includes a conservative treatment of intermediate volatility noble metals (IVNM). The modified treatment is more realistic compared to the highly conservative treatment using the Flibe radionuclide grouping structure and modeling of evaporative release in the MST methodology by adding a new IVNM group.

As stated in the SE for the approved MST methodology, the staff found it acceptable to group radionuclides by their chemical behavior in Flibe because this behavior impacts how the radionuclides are retained by Flibe. The TR differs from the MST methodology by placing elements that fall into multiple groups based on the Flibe oxidation-reduction (redox) potential in the most conservative grouping (i.e., resulting in more evaporative release to the gas phase). The MHA analysis in the TR accomplishes this for each element that may fall into multiple groups by comparing the vapor pressure of the representative species for the candidate radionuclide group to the product of the vapor pressure for cesium fluoride (CsF) multiplied by the concentration of the element in Flibe. Certain salt-soluble fluorides were also re-grouped based on data collected during the operation of the Kairos engineering test unit (ETU).

TR equation 2.35 provides the relationship for natural convection mass transfer between Flibe and the cover gas. The staff confirmed during the audit that the equation is an empirical correlation that Kairos is conducting tests to validate the correlation as described in limitation and condition 11 of the MST methodology TR SE. The staff will review the test results and derivation of this correlation when the MST or safety analysis methodologies are implemented.

Based on its review, the staff finds that the Flibe radionuclide grouping structure, including the representative species and applied release fraction for each group, is acceptable because it is based on well-established sources of thermodynamic data (references 13 through 16 in the TR) to derive vapor pressure correlations and determine uncertainties for the correlations that result in conservative modeling of the release from Flibe for radionuclides other than the salt-soluble fluorides. The staff confirmed in the audit both the information supporting the new IVNM grouping, including the natural convection mass transfer relationship between Flibe and cover gas, and the assumptions that lead to conservative modeling of radionuclide transport in Flibe. The staff also finds that the IVNM group is acceptable because the grouping leads to conservative radionuclide transport in Flibe. This is because the representative transport species (indium) for the IVNM group has a higher vapor pressure than all elements in the group and the relationship between vapor pressure and temperature includes a factor to account for uncertainty which is taken from a widely-used reference.

TR section 2.2.3.3 provides information to aid in confirming that the radionuclide concentrations in Flibe are consistent with the assumption in the MST methodology of a dilute solution as an initial condition. This information will be used by an applicant using the MHA analysis. As stated in the staff's SE for the MST methodology, the MST methodology assumes dilute solutions to minimize certain chemical interactions that could increase the vaporization of Flibe and radionuclide species and allows for the use of certain simplifying assumptions related to the retention of radionuclides in the molten salt. See also limitation and condition 5 in the MST methodology TR SE. TR section 2.2.3.3 also provides a method to show that the concentration of salt soluble fluorides is below the solubility limit by comparing the total concentration of the salt soluble fluorides to the solubility limit for the least soluble trivalent fluoride.

The staff evaluated TR information, including the use of cited references, to support the confirmation of dilute solution and solubility. Through the audit, the staff also confirmed the basis of the information in TR section 2.2.3.3 and its applicability to Flibe. The selection of the dilute solution concentration limit was based on solute-solute interaction data for a molten salt which is not Flibe but is justified as an appropriate surrogate based on the independence of the interaction from the solvent, chemical similarity of the salt to Flibe, and anticipation of a large margin between any individual impurity concentration and the limit. The staff notes that the TR information on the limit of solubility in Flibe for the salt soluble fluorides is based on solubility data for the least soluble trivalent fluoride. Therefore, based on its review, the staff finds that the information in TR section 2.2.3.3 is acceptable for use when confirming solubility and dilute solution to address limitation and condition 5 of the MST methodology TR SE.

TR section 2.2.3.4 addresses MST methodology TR SE limitation and condition 2 that the user of the MST methodology provide justification of thermodynamic data and associated vapor pressure correlations of representative species for the radionuclide groups. TR section 2.2.3.4 describes the basis for determining the vapor pressure correlations for the representative species and provides the resulting correlations for all the Flibe radionuclide transport groups used in the MHA. The staff evaluated the information in TR section 2.2.3.4 and audited information supporting the TR description.

For radionuclides other than the salt-soluble fluorides, the staff finds that the justification of the representative vapor pressure acceptable because the TR used well-established sources of thermodynamic data that can be applied to the KP-FHR operating conditions to derive vapor pressure correlations and determine uncertainties for these correlations that appear to be conservative. This includes the new IVNM grouping, which the staff still finds conservative, as previously discussed in this section of this SE.

The staff notes that confirmatory testing to validate the assumption of ideal vaporization behavior for salt-soluble fluorides as described in limitation and condition 11 in the MST methodology TR SE remains to be addressed by the user of the MST methodology, including through use of the safety analysis methodology.

2.3 Transport from Structural Materials

TR section 2.2.4 includes a description of the release of tritium, which is the same as in the Hermes and Hermes 2 PSARs. However, TR section 2.2.4.2 documents changes from the Ar-41 release modeling in the PSARs to include the immediate release of all Ar-41 dissolved in the Flibe, entrained in bubbles, and contained in the cover gas, and a conservative solubility-limited modeling of Ar-41 release from the reflector and fuel graphite pores to the Flibe that occurs over time. At 12 hours into the transient, any remaining Ar-41 is puff released out of the system.

The staff evaluated the updated Ar-41 release model, including an audit of an example analysis to aid in understanding the use of the solubility-limited pore release model described in TR section 2.2.1.1. The staff's evaluation of the Ar-41 pore release model finds it is acceptable, as discussed above in section 2.1 of this SE.

2.4 Release Pathway and Isotopic Screening Criteria

TR section 2.2.5 provides information to more succinctly address the MST methodology *de minimis* pathway screening based on the assessment of doses prior to the screening evaluation, such that a detailed computer calculation using RADTRAD is not required. Specifically, this new method involves assessment of isotopic releases for a given pathway to determine whether the pathway can be screened out of a detailed consequence modeling based on the very low offsite dose contribution from the pathway. This same process is used to identify significant dose contributors for the pathways that are not screened out of the consequence analysis and to reduce the number of isotopes included for RADTRAD calculation. The TR screening method is based on the method approved in the MST methodology TR, simplified to use a single absolute dose criterion of a pathway dose at the EAB of 0.001 rem total effective dose equivalent (TEDE).

The staff evaluated the more succinct screening process based on the previously approved method in the MST methodology and finds that the process achieves the same goal to determine that the MAR in the pathway and potential dose contribution would not affect the offsite dose result. The staff finds that the revised *de minimis* screening described in TR section 2.3.5 is acceptable because it assures that a comprehensive list of sources of MAR and release pathways (including those that are not likely to contribute more than a small fraction of the total offsite dose results) are included in the analysis with modeling assumptions consistent with their relative importance.

2.5 Control Room Dose Consequences

TR section 2.2.6 describes the radionuclide transport within the buildings and atmospheric transport, including calculation of offsite and control room dose consequences consistent with the approved MST methodology for calculating dose consequences for design basis accidents and control room habitability. The TR provides additional information to ensure that the control room atmospheric dispersion is modeled conservatively. The conservatism in the safety analysis methodology with respect to calculating control room dose consequences also includes not crediting filtration of the control room air to reduce the radionuclide concentration and not modeling the effect of shielding for reducing direct radiation exposure. Because the safety analysis methodology does not specifically model control room shielding or a potential filtration system, the staff will evaluate the control room dose analysis performed to show that KP-FHR PDC 19 control room radiological habitability criteria are met during the review of a future licensing action that uses the safety analysis methodology. Based on the staff's experience with modeling atmospheric dispersion for control room habitability analyses, the staff agrees that the information provided in the TR will result in conservative control room atmospheric dispersion modeling and resulting control room doses. Therefore, the staff finds that the methods described for the transport in the gas space and calculation of offsite and control room dose consequences are acceptable.

3. Safety Analysis Methodology Requirements

Element 1 of the RG 1.203 EMDAP identifies the following steps:

1. Specify Analysis Purpose, Transient Class, and Power Plant Class
2. Specify FOMs
3. Identify Systems, Components, Phases, Geometries, Fields, and Processes That Must Be Modeled
4. Identify and Rank Key Phenomena and Processes

Consistent with the guidance in EMDAP Element 1, section 3 of the TR describes the KP-FHR test reactor postulated events modeled by the safety analysis methodology, specifies FOMs and acceptance criteria, and identifies important SSCs and associated phenomena and processes that need to be modeled to calculate the target FOMs.

Consistent with the guidance in NUREG-1537, part 1, chapter 13, section 3 of the TR identifies and categorizes the postulated events in the KP-FHR test reactor. The events are categorized based on similarity of their characteristics into the following six categories:

- Increase in heat removal
- Decrease in heat removal
- Loss of forced circulation
- Reactivity-initiated event
- Salt spills
- Pebble handling and storage system (PHSS) malfunction
- Radioactive release from a subsystem or component

The frequency of event occurrence is not considered in the categorization of events. The staff finds that this is acceptable because all the events are considered equally important and treated similarly for the conservative modeling in the safety analysis methodology.

Limitation 1 in TR section 6.2 clarifies that the safety analysis methodology presented in the TR is applicable to the postulated event categories described in TR section 3.2 and the safety analysis methodology for event categories not included in TR section 3.2 will be provided in future licensing submittals. The staff includes this limitation through **Limitation and Condition 1**, to ensure that future licensing submittals that include any new event category will address the applicable steps in EMDAP consistent with the safety analysis methodology presented in this TR. In addition, as highlighted by limitation 5 in TR section 6.2, any new design features not considered in TR section 1.1 should be addressed, consistent with the guidance in EMDAP, in a future licensing submittal; the NRC staff include this limitation through **Limitation and Condition 1** of this SE.

3.1 Postulated Event Description

TR section 3.2 describes progression of a typical event in each postulated event category and identifies the potential limiting event for each postulated event category. TR section 5 presents a deterministic approach for identification of the limiting event by performing sensitivity calculations. The staff's evaluation of this approach for determining the limiting event is presented in SE section 3.5 below.

The staff reviewed the progression of a typical event in each postulated event category and observed that the initial KP-FHR system response to a generic postulated initiating event is generally characterized by the changes in core flow and temperature and the resultant reactivity feedback that affects the core fission power. The reactivity feedback is dominated by Doppler, moderator, or coolant temperature feedback depending on the initiating event. In general, the inherent negative reactivity feedback prevents uncontrolled rise of core power and overheating fuel during this initial transient phase before the activation of the reactor protection system (RPS). The RPS response triggered by the activation safety signals results in the insertion of shutdown elements and tripping of the primary salt pump (PSP). The heat rejection blower as well as the PHSS are also tripped on activation of RPS. The safety signals that can initiate the RPS response include high coolant temperature, low coolant level, high core power, high power rate, or low PHSS pressure. The activation of RPS results in the termination of core fission power and a transition to the decay heat generation mode. The loss of primary heat transport system (PHTS) circulation due to PSP trip (or due to initiating event) causes in-vessel flow to transition to a natural circulation model. Although the decay heat removal system (DHRS) is always active, initial core decay heat exceeds the heat removed by DHRS, leading to a slow heat up of the core and reactor system. Eventually, the DHRS heat removal exceeds the core decay heat, and the event progression transitions into the long-term continuous cooldown phase. TR section 3.2 clarifies that the analysis of Flibe freezing conditions for the long-term phase is not included in the safety analysis methodology and will be addressed in a future licensing submittal. This limitation is identified as limitation 2 in TR section 6.2. The staff includes this limitation through **Limitation and Condition 1** of this SE.

A salt spill event, as described in TR section 3.2.5, is initiated by a breach in the Flibe-carrying SSCs connected to the reactor vessel (e.g., PHTS piping such as hot and cold legs, heat rejection radiator tubing). As described in TR section 1.1.3, there are two anti-siphon break points in the PHTS: the PSP siphon break point for the hot leg and the reactor vessel siphon break point for the cold leg. These passive antisiphon features limit the amount of spilled coolant salt to the level above the elevation of natural circulation path (NCP). Since the RPS would activate either due to the low coolant level or high coolant temperature actuation signals, an adequate amount of salt is preserved in the reactor vessel to keep the NCP submerged for continuous removal of decay heat by the DHRS. Following the activation of RPS, the progression of the salt spill event is consistent with the generic event description provided in the previous paragraph of this SE.

The PHSS malfunction event, as described in TR section 3.2.6, addresses the event initiated by the break in a PHSS transfer line. This event is detected by the RPS through the PHSS low pressure actuation signal. The event progression after the activation of RPS is consistent with the generic event description provided above. TR section 3.2.6 indicates that the other PHSS malfunctions such as loss of PHSS cooling and mechanical damage to a pebble in the PHSS line are assumed to be mitigated by design and the safety analysis methodology is only applicable to the PHSS malfunction due to a transfer line break. The TR does not identify any phenomena unique to the PHSS malfunction event in TR section 3.5 or describe any unique KP-SAM modeling approach for the PHSS malfunction event in TR sections 4 and 5. The staff finds that the level of detail provided in the TR for the PHSS transfer line break event are acceptable because the event is mitigated by a dedicated RPS actuation signal that is based on the low pressure in PHSS. However, information is currently not available on design features to mitigate other PHSS malfunctions and to identify the limiting event for this event category consistent with the guidance in NUREG-1537, part 1, chapter 13. Therefore, the staff impose **Limitation and Condition 2**, to limit the applicability of the safety analysis methodology to a PHSS malfunction initiated by a break in transfer line.

TR sections 3.2.5 and 3.2.6 describe the potential for air to get entrained into the PHTS, reactor vessel, or PHSS through the break in the event of salt spill or PHSS transfer line break. The interaction of air with Flibe can generate volatile products. Furthermore, ingress of air into the reactor vessel and PHSS can lead to the generation of flammable gas due to oxidation of un-submerged structural graphite and carbon pebbles. However, limitation 4 in TR section 6.2 clarifies that the safety analysis methodology does not address the generation of flammable gas due to graphite oxidation in reactor vessel following an air ingress scenario. The staff includes this limitation through **Limitation and Condition 1**.

The postulated event category described in TR section 3.2.7 accounts for the release of radioactive material due to failure of a subsystem or component which contains radioactive material outside of the functional containment (i.e., TRISO particle and Flibe coolant). The safety analysis methodology assumes the MAR in one or multiple affected subsystems and components is fully released at the start of event. The staff finds that the assumption of release of all the MAR at the start of the event is conservative with respect to estimating dose and is, therefore, acceptable.

Based on its review, the staff finds that the description of event progression and the expected system response described in TR section 3.2 are consistent with the example calculations presented in TR appendix A. In addition, the staff finds that postulated event description, consistent with the guidance in EMDAP element 1, provides a reasonable level of detail needed to evaluate the safety analysis methodology. Furthermore, the staff finds that the postulated event categorization presented in the TR is consistent with the requirements of NUREG-1537, part 1, chapter 13.

3.2 Figures of Merit

The important FOMs for the safety analysis methodology are related to the dose criteria for test reactor siting, as required by 10 CFR 100.11. The regulatory criteria are stated in terms of whole body and thyroid from radioiodine doses to an individual at the EAB for the initial two hours and at the LPZ for the duration of the passage of the plume from an accident. In addition, KP-FHR PDC 19 provides a control room dose criterion stated in terms of TEDE in the control room for the duration of the accident.

The MHA in the safety analysis methodology is explicitly evaluated to provide dose results that demonstrate compliance with each of these accident dose criteria for the application. In lieu of explicitly demonstrating that the doses for each of the postulated events meet the regulatory dose criteria FOMs stated above, the safety analysis methodology demonstrates that the postulated event dose pathways are bounded by the MHA. The primary dose FOM for the safety analysis methodology is the MHA 30-day TEDE at the EAB which is an aggregated dose that includes impacts from whole body, thyroid, and other organ doses. The 30-day TEDE at the EAB for each postulated event is then shown to be bounded by the MHA 30-day TEDE at the EAB. This is accomplished by evaluating the postulated events to confirm that the release pathways not considered in the dose consequence analysis for the event do not occur and demonstrating that the dose consequences of the postulated events are below those of the MHA.

To ensure that each postulated event analysis compared to the 30-day dose FOM is also relevant for all potential source term release timings and to the regulatory dose criteria exposure timing in 10 CFR 100.11, the postulated events analyses for each postulated event group also estimates the whole body and thyroid from radioiodine doses for the first two hours at the EAB and duration of the passage of the plume at the LPZ. Therefore, the staff finds that the use of 30-day TEDE at the EAB to compare to the dose FOM instead of whole body and thyroid from radioiodine doses is reasonable for the purpose of demonstrating that the MHA is bounding for other postulated event groups.

TR sections 3.3.1 through 3.3.5 describes the additional surrogate safety analysis methodology FOMs evaluated to confirm that the postulated events do not include releases not modeled in the MHA. These surrogate FOMs are summarized in table 1 of this SE. For each FOM, table 1 summarizes acceptance criteria, justification or basis, evaluation approach, and the postulated event category for which the FOM is evaluated. Table 1 also shows the 30-day TEDE dose FOM which is evaluated for each postulated event category to demonstrate that the dose consequences of the postulated events are below those of the MHA.

As shown in table 1, peak TRISO silicon carbide (SiC) layer temperature and fuel failure fraction FOMs are evaluated using the KP-BISON code, which is described in TR KP-TR-010-NP-A, “KP-FHR Fuel Performance Methodology” (ML22125A278). The peak pebble power and peak pebble surface temperature calculated using the KP-SAM hot-pebble factor (HPF) EM described in section 4.2.4 of KP-TR-020, Revision 1, and evaluated in this SE in section 4.1 are provided as input to KP-BISON. The KP-BISON methodology for calculation of the peak TRISO SiC layer temperature and fuel failure fraction FOMs is not presented in this TR; therefore, the staff impose **Limitation and Condition 3** that clarifies that the KP-BISON methodology for calculation of these FOMs is not reviewed and approved in this SE and the acceptable justification for the use of KP-BISON for the calculation of these FOMs must be provided with the licensing application referencing this TR.

TR sections 3.3.4 and 3.3.5 clarify that the acceptance criteria for the peak vessel temperature and peak structural graphite temperature FOMs are consistent with the material qualification data for the KP-FHR test reactor. The staff impose **Limitation and Condition 4(a)** to require an application referencing this TR to provide supporting material qualification data for the review. A conservative approach for the calculation of these FOMs is presented in TR section 4.2.4 and evaluated in section 4.1 of this SE.

TR section 3.3.4 clarifies that the peak vessel temperature FOM is also used as surrogate FOM for the maximum temperature of metallic materials outside of the pebble bed core (e.g., core barrel and hold-down structure). However, as identified by limitation 6 in TR section 6.2, the evaluation of peak temperature of metallic materials inside the pebble bed (e.g. shutdown elements) is not within the scope of the TR methodology. The staff includes this limitation through **Limitation and Condition 1**.

Based on its review, the staff finds that the TR, consistent with the guidance in EMDAP Element 1, provides an adequate description of the primary and surrogate FOMs and identifies appropriate acceptance criteria for the safety analysis methodology. Furthermore, the staff finds that the proposed surrogate FOMs are suitable for supporting the development of the KP-SAM EM for postulated event analysis.

Approach for Calculation of Dose FOM

TR section 3.3.6.1 describes the safety analysis methodology approach for evaluation of the dose FOM for three types of postulated events: primary system postulated events with MHA release pathways, non-intact postulated events with MHA and non-MHA release pathways, and releases from subsystem or component postulated events. As discussed in TR section 3.3.6.1 and summarized in table 2 of this SE, for the primary system postulated events 30-day TEDE at the EAB is calculated using the MHA release pathways (i.e., releases from the intact primary system) and bounding temperature-time and flowrate-time curves for the following parameters:

- Maximum kernel temperature (also conservatively applied to TRISO layers)
- Maximum reflector temperature
- Maximum pebble surface temperature
- Flibe-cover gas interface temperature
- Core mass flowrate
- Bypass mass flowrate

The postulated event categories considered for this approach are the events with the MHA release pathways from the primary system including the increase and decrease in heat removal events, loss of forced circulation, reactivity-initiated events, and salt spill events. The KP-SAM transient calculations for the limiting events for these event categories are used to confirm the bounding nature of the selected parameter curves. The limiting event for each category is determined using the sensitivity calculations proposed in TR section 5 and the conservative bounding values for the temperature related FOMs including peak TRISO temperature, peak vessel temperature, and peak reflector temperature calculated using the approach presented in TR section 4.2.4. However, the bounding nature for the dose analysis input parameter curves is confirmed using the nominal maximum parameter values. TR section 3.3.6.1 further describes the approach used for calculation of bounding parameter curves from the KP-SAM calculation. The staff finds that the use of KP-SAM calculated nominal maximum parameter values for confirming the bounding parameter curves for dose analysis is acceptable because the nominal maximum temperatures for kernel, reflector, and pebble surface are expected to be higher than the average temperatures for these materials during the transient.

TR section 3.3.6.2 describes the safety analysis methodology approach for the calculation of dose FOM in non-intact postulated events. For the non-intact primary system postulated event with non-MHA release pathways the sum of the 30-day TEDE at the EAB for all non-MHA release pathways for the event is added to the 30-day TEDE at the EAB from the bounding primary system postulated event (described in TR section 3.3.6.1) to give a total consequence for the event to compare against the MHA 30-day TEDE at the EAB dose FOM. The calculation of the 30-day TEDE at the EAB for non-MHA release pathways in salt spill events and PHSS malfunction postulated events is described in TR section 5 and evaluated in SE section 5.2.

For the releases of subsystem and component event category, as described in TR section 3.3.6.3, the MAR in one or more subsystem or components is assumed to be fully released to calculate the 30-day TEDE at the EAB for comparison against the MHA 30-day TEDE at the EAB dose FOM.

Based on its review, the staff finds that the approaches described in TR section 3.3.6 consider the potential radiological releases for postulated events and differences from the scenario modeled in the MHA analysis to demonstrate that the MHA consequences are bounding for the test reactor postulated events. Therefore, the staff finds that the FOMs based on these approaches also demonstrate that the MHA is bounding of the test reactor postulated events.

Table 1 Safety Analysis Methodology Figures of Merit

FOM	**Acceptance Criteria**	**Basis or Justification**	**Calculation Approach**	**Applicable Event Categories**
Peak TRISO SiC temperature	1600 °C	Ensures that SiC layer failures do not occur in transient conditions and transient failure fraction does not deviate from the steady state failure fraction assumed in MHA dose analysis. Ensures that the fuel remains within the qualification envelope described in TR KP-TR-011-NP-A, “Fuel Qualification Methodology for the Kairos Power Fluoride Salt-	Calculated using KP-BISON with input of peak pebble power and peak pebble surface temperature calculated using KP-SAM HPF approach.	Increase or decrease in heat removal, Loss of forced circulation, Reactivity-initiated event, and Salt spills

		Cooled High Temperature Reactor (KP-FHR)," (ML23089A398).		
Fuel failure fraction	Steady state failure fraction assumed in MHA dose analysis	Ensures that transient failure fraction does not deviate from the steady state particle configuration fraction. Ensures that fuel performance is consistent with that assumed in the MHA dose analysis.		
Peak vessel temperature	750 °C	Consistent with material qualification for KP-FHR test reactor. Ensures that coolable geometry is maintained and vessel failure do not occur as assumed in the MHA dose analysis.	Calculated using KP-SAM peak vessel temperature and peak reflector temperature surrogate described in TR section 4.2.4 and evaluated in section this SE section 3.3.	
Peak structural graphite temperature	950 °C	Consistent with material qualification for KP-FHR test reactor. Ensures that structural integrity of graphite is maintained to permit the insertion of shutdown element and to facilitate natural circulated needed to maintain the coolable geometry as assumed in the MHA dose analysis.		
Energy deposition pulse width	Greater than 1 second	Ensures that fuel failures not considered in MHA do not occur in reactivity-initiated events. As the TRISO fuel constant is small, the selected acceptance criteria assures that energy deposited in TRISO fuel is dissipated without significant heat up.	Time from initiation of event to time of maximum power in KP-SAM simulation.	Reactivity-initiated event
30-day TEDE dose at EAB	See table 2 of this SE for acceptance criteria and approach for calculation of this FOM.			All postulated event categories

Table 2 Safety Analysis Methodology Approach for Calculation of Dose Figure of Merit

Postulated event	**Acceptance criteria**	**Basis or Justification**	**Evaluation Approach**
Primary system postulated events	30-day TEDE at EAB < (less than) MHA 30-day TEDE at EAB	TEDE is an aggregate dose FOM that includes impacts to the whole body, thyroid, and other organs	Bounding nominal parameter curves calculated using KP-SAM EM used to calculate 30-day TEDE at EAB
Non-intact postulated events (salt spills, PHSS malfunction)	30-day TEDE and 30-day TEDE at EAB from all non-MHA release pathways < (less than) MHA 30-day TEDE at EAB		Sum of 30-day TEDE doses at EAB calculated for all non-MHA pathways added to the 30-day TEDE at EAB for the bounding primary system postulated event

Release from subsystem and component	30-day TEDE at EAB for bounding sequence < (less than) MHA 30-day TEDE at EAB	No releases from primary system for this event.	Only release pathways from the postulated events are directly compared to MHA dose.

3.3 Phenomena Identification and Ranking Table

The importance rankings and knowledge levels assigned to phenomena in the PIRT guide the development of the EM (Element 3 of EMDAP), validation base (Element 2 of EMDAP), and applicability evaluation (Element 4 of EMDAP). For the KP-SAM EM presented in the TR, Kairos referenced the thermal fluids PIRT and the reactivity insertion PIRT that were developed by Kairos for the KP-FHR. The thermal fluids PIRT identified thermal-hydraulic phenomena that are relevant to all the postulated event categories except for the radioactive release from subsystem and component event category. The reactivity insertion PIRT addressed the specific phenomena in the reactivity-initiated event and the increase in heat removal events.

Kairos also identified additional PIRTs (i.e., neutronics PIRT, fuel PIRT, and structural PIRT) that have an indirect impact on the safety analysis methodology presented in the TR. The neutronics PIRT is used in the core design and analysis methodology described in TR KP-TR-024-NP, “KP-FHR Core Design and Analysis Methodology” (ML25168A340), which provides the nuclear parameter inputs to the KP-SAM point kinetics model used in the TR to calculate fission power. The fuel PIRT is used for the development of KP-BISON based fuel performance methodology in KP-TR-010-NP-A. KP-BISON is used to provide input to the source term analysis and to assess the two FOMs (i.e., peak SiC layer temperature and fuel failure fraction) of the safety analysis methodology. The structural materials PIRT is used in TR KP-TR-013-NP-A, “Metallic Materials Qualification for the Kairos Power Fluoride Salt-Cooled, High Temperature Reactor,” (ML23102A179), which has an impact on the acceptance criteria for the structural materials in the safety analysis methodology.

Table 3-1 of the TR summarizes the high and medium importance phenomena from the thermal fluids PIRT and how they are addressed by the KP-SAM EM. The thermal fluids PIRT identifies the phenomena in the following SSCs: **[[**

]] The staff’s evaluation of the approach used to model these PIRT phenomena and associated modeling assumptions is provided in SE section 4.3.

Although most of the thermal fluids PIRT phenomena apply to all the event categories analyzed using the KP-SAM model, some phenomena are unique to specific event categories, such as the salt spill event (e.g., **[[** **]]**). The PIRT also initially identified some phenomena as medium or high importance but later downgraded them to low importance based on insights from the sensitivity analysis, such as **[[** **]]** The staff finds that this iterative approach for updating the PIRT is acceptable because it reduces the subjectivity in the PIRT and increases the focus on important phenomena.

The thermal fluids PIRT identifies phenomena in the **[[**

]] The PIRT justifies this based on the modeling approach for DHRS that involves the use of experimental data to derive the bounding high and low DHRS performance curves. Further evaluation of the DHRS modeling approach is provided in SE section 4. The thermal fluids PIRT identified some phenomena that are not represented in the KP-SAM model. These include the phenomena related to the modeling of reactor physics (e.g., **[[** **]]**) and the pebble and TRISO fuel properties and heat transfer (e.g., **[[**
]]). These phenomena are modeled by the other methodologies described in KP-TR-024-NP or KP-TR-010-NP-A.

Table 3-2 of the TR summarizes the high and medium ranked phenomena in the reactivity PIRT. The reactivity PIRT identifies the phenomena in the following categories: **[[**

]] The phenomena presented in this PIRT are either addressed in the thermal fluids PIRT or captured in the KP-TR-010-NP-A methodology used for the calculation of two FOMs as discussed in SE section 3.2 or accounted by the KP-TR-024-NP methodology used to generate conservative inputs for the point kinetics model in KP-SAM.

Table 3-3 of the TR summarizes the new high and medium importance ranked source term phenomena that were not previously identified in KP-TR-012-NP-A and the phenomena for which the importance ranking was upgraded. The staff finds that the new source term PIRT information is relevant to the more detailed radionuclide transport modeling in the primary system, for salt spills, and for releases from the PHSS as described in TR sections 2.2, 5.5.3, and 5.6.2, respectively. The staff's evaluation of the approach used to model these PIRT phenomena and associated modeling assumptions is provided in SE section 2 for the models in TR section 2.2, and in SE section 5.2 for the event-specific source term methods discussed in TR sections 5.5.3 and 5.6.2.

The PIRTs presented in the TR **[[** **]]** of the system consistent with EMDAP Step 3. The PIRTs also **[[**
]] as recommended by EMDAP Step 4 as the different SSCs and the associated phenomena might have a different importance level during the multiple phases of the accident or transient. For example, the phenomena that are important during the early phase of transient before the activation of RPS may have a different importance level after the RPS activation or in the long -term phase. However, the staff finds that this simplified approach is acceptable for the test reactor safety analysis methodology development. This is because the objective of the safety analysis methodology is to demonstrate that the predicted temperature-time or flow-time for the limiting events are bound by the parametric curves provided as input to the MHA analysis for the calculation of limiting dose.

4. KP-SAM Code, Base Model, and Validations

TR section 4 addresses EMDAP Element 2 and Element 3 for the KP-SAM EM.

EMDAP steps under Element 2 are:

- Step 5, "Specify Objectives for Assessment Base"
- Step 6, "Perform Scaling Analysis and Identify Similarity Criteria"

- Step 7, “Identify Existing Data and/or Perform Integral Effects Tests (IETs) and Separate Effects Tests (SETs) to Complete the Database”
- Step 8, “Evaluate Effects of IET Distortions and SET Scaleup Capability”
- Step 9, “Determine Experimental Uncertainties as Appropriate”

These steps are addressed, in part, in TR sections 4.2 and evaluated in SE section 4.3.

EMDAP steps under Element 3 are:

- Step 10, “Establish an Evaluation Model Development Plan”
- Step 11, “Establish Evaluation Model Structure”
- Step 12, “Develop or Incorporate Closure Models”

These steps are addressed in TR sections 4.1 and 4.2. The staff’s evaluation of KP-SAM EM documentation, quality assurance, and configuration control for EMDAP Step 10 is presented in SE section 4.4. The staff’s evaluation of KP-SAM code and base model for EMDAP steps 11 and 12 is presented in SE sections 4.1 and 4.2.

4.1 KP-SAM Code Description

The KP-SAM code is based upon the System Analysis Module (SAM) code, which was developed by Argonne National Laboratory (ANL) under the Department of Energy’s Nuclear Energy Advanced Modeling and Simulation (NEAMS) workbench as a generic system-level safety analysis tool for advanced non-LWRs. It is capable of modeling thermal-hydraulics and neutronics phenomena and control system behavior for integral analysis of transients and accidents in advanced reactors. The TR identifies the following major developments from SAM to KP-SAM:

- a description of the physics and system component behaviors;
- specific numerical methods such as continuous finite element method stabilizing methods; and
- addition of specific functions for the systems code.

According to section 4.1.1 of the TR, KP-SAM, like SAM, has two types of physics models: field equations and closure models. Field equations are solved to transport quantities of interest in 0-dimensional (D), 1-D, or 2-D domains. However, the KP-SAM EM presented in the TR does not use 2-D flow-field equations; it only uses 1-D single-phase flow-field equations for mass, momentum, and energy conservation along the flow direction and their primary variables are pressure, velocity and temperature. Heat structures in KP-SAM model the heat conduction inside the solids and permit the modeling of heat transfer at the interfaces between solid and fluid components. Heat structures are represented by 1-D or 2-D heat conduction in Cartesian or cylindrical coordinates. A 1-D spherical heat conduction model is also developed for pebble bed simulation. KP-SAM also includes generic closure models and correlations. Closure models and correlations specific to the KP-FHR design and their validations are discussed in TR section 4.2 and evaluated in SE section 4.1.2. These include, for example, correlations for flow and heat transfer in the pebble bed core.

The KP-SAM control system is used to perform evaluation of algebraic and other simple equations; the trip system is used to perform the evaluation of logical statements. The main execution of individual control/trip units is set at the end of each time step.

KP-SAM uses a continuous finite element method formulation for the spatial discretization of the field equations, with the code formulated such that the numerical method orders are controlled through user inputs. For fluid models, a spatial stabilization method is implemented to suppress checkerboard-type spatial oscillations that manifest when using continuous finite element methods to solve advection dominated problems.

As described below, the staff reviewed and evaluated the KP-SAM code capabilities, limitations, closure models and validations as described in this TR and the KP-SAM theory manual KP-RPT-000231, “KP-SAM Theory Manual” (ML24156A166), and user guide KP-RPT-000226, “KP-SAM Users Guide,” (ML24156A167) to assess the adequacy of KP-SAM for the calculation of safety analysis methodology FOMs.

4.1.1 KP-SAM Code Limitations

TR section 4.2.4 describes the KP-SAM limitations and the safety analysis methodology approach to address these limitations. The following KP-SAM limitations are addressed by the safety analysis methodology:

- Use of 1-D fluid flow and 2-D heat transfer approach for modeling of flow and temperature distribution in reactor (core, reflector, downcomer, and vessel wall).
- Use of uniform porosity for pebble bed in core.
- Use of 1-D power distribution in core.
- Neglecting radiative heat transfer in reactor.
- No model to capture gamma and neutron heating of in-vessel heat structures (i.e., reflector and metallics). All energy from gamma and neutron heating is assumed to deposit in fuel pebbles.

These limitations, summarized in TR table 4-4, result in limited capability for KP-SAM to model 3-D temperature distributions in the core, reflector, and vessel wall. The safety analysis methodology uses conservative modeling methods or factors to account for these limitations for the calculation of key FOMs: an HPF, a peak vessel temperature surrogate, and a peak reflector temperature surrogate.

Hot Pebble Factor

The HPF, described in TR section 4.2.4.1, is used for the conservative calculation of pebble surface temperature which is then provided as input to KP-BISON for the calculation of the peak TRISO layer temperature FOM. It accounts for the impact of 3-D flow and power distribution in the core and uncertainties in pebble bed heat transfer correlations and pebble diameter.

The HPF is based on the hot channel factor approach used in nuclear reactor safety analysis to account for impact of different uncertainties on the calculation of safety parameters such as peak fuel temperature (References 1 and 2). It uses vertical semi-statistical methods to combine deterministic and statistical uncertainties. The HPF in the TR calculates conservative bounding factors as multipliers for the coolant temperature rise and the maximum pebble film temperature rise (i.e., maximum temperature difference between the pebble surface temperature and the coolant temperature). The maximum pebble surface temperature is calculated using TR equation 4.8 from the bounding values of the coolant temperature rise and the maximum pebble film temperature rise.

The bounding coolant temperature rise is calculated by multiplying the KP-SAM calculated maximum coolant temperature rise by a direct subfactor that accounts for the impact of radial and azimuthal power distribution. The upper bound radial and azimuthal peaking factor value calculated from the static core design analysis is used for this subfactor. The 2-D static core design analysis is used for events with no azimuthal power variation to calculate the radial peaking factor. The 3-D static core design analysis is used if the transient induces radial as well as azimuthal power distribution perturbation (e.g., single control rod withdrawal). The axial power distribution is accounted for in the coolant temperature rise calculated by KP-SAM. The TR justifies the use of static core analysis for the calculation of peaking factors because the flux distribution in a transient spatial kinetics calculation would be flatter due to reactivity feedback effects. The staff finds that the use of 2-D or 3-D static core analysis for the calculation of upper bound radial and azimuthal peaking factor is acceptable because it results in conservative values for the peaking factor. The example calculation A.5 in appendix A of the TR estimates a value of **[[ ]]** for the radial and azimuthal peaking factor. The HPF does not account for the impact of radial flow distribution on the coolant temperature rise. Kairos clarified in audit discussions that the pebble bed induced mixing substantially reduces the impact of radial flow distribution on the coolant temperature rise. The staff finds this explanation reasonable.

The bounding maximum film temperature rise is calculated by multiplying the KP-SAM calculated maximum film temperature rise by two direct and two statistical subfactors that are combined using TR equation 4.6. The direct subfactors account for the impact of distributions in pebble power and local flow on the calculated pebble film temperature. The impact of pebble power distribution is accounted for using a pebble peaking factor which is defined as peak pebble power relative to average core power. The same approach described for the radial and azimuthal peaking factor is used for the calculation of pebble peaking factor (i.e., use of 2-D or 3-D static core analysis). The example calculation A.5 in appendix A of the TR estimates a value of **[[ ]]** for the pebble peaking factor. The direct subfactor that accounts for the impact of local flow distribution (flow penalty subfactor) is calculated using TR equation 4.12 with the support of the computational fluid dynamics (CFD) **[[ ]]** packed bed model that was described in KP-TR-024-NP. The TR states that the single flow penally subfactor for the symmetric and asymmetric events is adequate because the preliminary analysis shows that the **[[ ]]**. The TR states that the bounding value for this factor will be calculated over a range of core flow rates or Reynolds numbers. The TR further states that the uncertainties in the CFD **[[ ]]** model will be considered, and additional discretionary conservatism will be applied to account for reliance on the CFD calculation performed for **[[ ]]**.

The two statistical subfactors for the film temperature rise account for the uncertainties in pebble diameter and convective heat transfer correlation for the pebble surface heat transfer (Wakao correlation, Reference 3). TR equation 4.13 calculates the statistical subfactor for the pebble diameter uncertainty based on the Wakao correlation. The statistical subfactor for the **[[** **]]** is calculated based on comparison of **[[** **]]** correlation to the experimental data. The example calculation A.5 in appendix A of the TR estimates values of **[[** **]]** and **[[** **]]** for the pebble diameter and **[[** **]]** correlation statistical subfactors, respectively.

The HPF does not explicitly account for the impact of uncertainties associated with calculation of total core power, core and bypass flow split, neutronics parameters and calculated power response. Kairos clarified in audit discussion that these uncertainties are separately addressed in TR section 5 as a part of event-specific event specific biases and sensitivity analysis. Additionally, the HPF does not account for the KP-SAM model limitations related to the modeling of gamma and neutron heating and radiation heat transfer. The staff finds that this modeling limitation is acceptable because it leads to the deposition of all core energy into the coolant which results in a conservative estimation of peak pebble surface temperature.

The results of the example calculation A.5 in appendix A of the TR illustrate that the contributions from the two direct subfactors that account for the impact of power distribution (i.e., radial/azimuthal peaking factor and pebble peaking factor) and the statistical subfactor that accounts for the uncertainty in pebble surface heat transfer correlation dominate the calculation of peak pebble surface temperature using the HPF. Based on the review of TR section 4.2.4.1, the staff finds that the HPF adequately addresses all the identified KP-SAM limitations. However, the HPF is dependent on the 2-D or 3-D static core design analysis for the calculation of radial/azimuthal and pebble peaking factors. Furthermore, the HPF is also dependent on the use of the CFD **[[** **]]** model for the calculation of direct subfactor that accounts for the impact of local flow distribution (flow penalty factor). KP-TR-024-NP provides a methodology for the calculation of radial/azimuthal and pebble peaking factors. Also, it appears that the CFD **[[** **]]** model presented in KP-TR-024-NP for the validation of porous media model can be used for the calculation of local flow penalty subfactor. However, neither this TR nor KP-TR-024-NP provided any calculations demonstrating the implementation of 2-D or 3-D static core design analysis or the CFD **[[** **]]** calculations for the estimation of direct subfactors in the HPF for different event categories. Therefore, the staff is imposing **Limitation and Condition 4 (b)**, that requires test reactor applications referencing this TR to provide for staff review the supporting 2-D/3-D static core analysis and the CFD **[[** **]]** model results used for the calculations of peaking factors and local flow penalty factor.

<u>Peak Vessel Temperature Surrogate</u>

The peak vessel temperature surrogate, described in TR section 4.2.4.2, accounts for the impact of the KP-SAM limitations on the calculation of peak vessel temperature. The Flibe temperature at the outlet of the NCP is used as surrogate for the peak vessel temperature. This approach is conservative because the Flibe temperature at the outlet of the NCP is maximum within the system and it neglects the loss of Flibe energy in the NCP, downcomer, and by DHRS heat removal. Furthermore, as discussed earlier, the coolant temperature calculated in KP-SAM is also maximized due to the limitations on modeling of radiation heat transfer and gamma and neutron heating. The staff finds that the approach for the calculation of peak vessel temperature FOM acceptable because it is conservative.

Peak Reflector Temperature Surrogate

The peak reflector temperature surrogate, described in TR section 4.2.4.3, accounts for the impact of the KP-SAM limitations on the calculation of peak reflector temperature. TR equation 4.14 is used to calculate the peak reflector temperature. Similar to the peak vessel temperature surrogate, the maximum Flibe temperature is used but with the penalty factors to account for the impact flow and power distribution in core on the local reflector temperature. The radial peaking factor is calculated using a similar approach as described for the HPF approach for uniform and asymmetric power distribution events. The flow factor that accounts for the local flow reduction is calculated from the **[[ ]]** analysis. TR table 5-1 summarizes the different direct subfactors used for the HPF and peak reflector temperature surrogate calculation and the interface with KP-TR-024-NP. The staff finds that the approach proposed for the peak reflector temperature accounts for the identified KP-SAM limitations. However, **Limitation and Condition 4 (b)** is also applicable to the 2-D/3-D static core design analysis and the CFD **[[ ]]** model analysis performed to estimate the peak reflector temperature surrogate.

The staff finds that the TR and the KP-SAM code manuals provide adequate description of the code fundamental equations and numerical methods. Furthermore, the TR adequately describes the KP-SAM limitations and EM approach to address those limitations satisfying, in part, EMDAP step 11. The KP-SAM base model described in TR section 4.3 and the event-specific biases and sensitivity described in TR section 5 address the remaining scope of EMDAP step 11.

4.1.2 KP-SAM Closure Relations

The key closure models implemented in KP-SAM include the equation of state for Flibe, material properties for solids (such as graphite and metallic materials), heat transfer correlations, and wall friction correlations. In addition to the TR, the KP-SAM theory manual provides a detailed description of the closure models available in the code.

[[ ]] TR section 5 indicates that the Flibe thermal-physical properties used in the KP-SAM EM are biased to bound the uncertainties described in TR KP-TR-005-NP-A, "Reactor Coolant for the Kairos Power Fluoride Salt-Cooled High Temperature Reactor," (ML20219A591). Further evaluation of event-specific biases proposed for the Flibe properties is in SE section 5. **[[**

]]. Furthermore, as shown in **[[ ]]** and described in TR section 4.3.6, the thermal conductivity of the ET-10 reflector graphite material is biased in KP-SAM to a conservatively low value, corresponding to bounding irradiation temperature and fluence to maximize stored energy in reflector. **[[**

]] to reduce calculated margin to fuel safety limit (i.e., peak TRISO layer temperature). However, the final calculation of peak TRISO temperature FOM performed in KP-BISON uses thermal properties as function of temperature and irradiation. Therefore, based on this review, the staff finds that the basis for the equation of state for Flibe and the material properties used in the KP-SAM EM is acceptable.

KP-SAM theory manual chapters 17 and 18 describe the closing relations for heat transfer and wall friction available in KP-SAM. TR section 4.2.2 describes the key closing relations used to model the selected high importance ranked phenomena including: **[[ ]]**. TR section 4.2.2 also describes the SETs planned to validate these correlations. The evaluation of these proposed SET validations is provided in SE section 4.3.

TR section 4.2.2.1 describes the correlations selected to model the **[[ ]]**. The staff finds that the proposed correlation for the **[[ ]]** and will be validated against applicable SET data.

TR section 4.2.2.2 describes **[[ ]]**. The staff finds the use of the **[[ ]]** acceptable because it has a well-documented basis and validation is planned against applicable SET data as described in TR section 4.2.2.2.

TR section 4.2.2.3 describes **[[ ]]**. The staff finds the use of **[[ ]]** acceptable because it has a well-documented basis and validation is planned against applicable SET data as described in TR section 4.2.2.3.

TR section 4.3.7 describes the closing relation used **[[**

]].

Based on its review, the staff finds that the KP-SAM closure models are acceptable because the TR, KP-SAM user guide and theory manuals provide adequate information on the closure models to address EMDAP step 12.

4.2 KP-SAM Base Model

The KP-SAM base model described in TR section 4.3 is a best estimate model that can be adequately used for simulation of KP-FHR test reactor for postulated event analysis provided that the biases and assumptions required to perform conservative safety analysis, such as the HPFs, are applied. The vessel portion is modeled as axisymmetric around the axial direction (z-axis), with the fluid (Flibe) region modeled in 1-D and the solid regions modeled in 2-D (r-z). This modeling approximation is adequate for the test reactor since relevant flow and heat transfer conditions are expected to be axisymmetric, and power distribution asymmetries are captured by the HPF.

The core flow, including the cylindrical (CYL), converging (CON), and diverging (DIV) regions, is modeled in 1-D with nodalization only along the axial direction (i.e., with no radial or azimuthal nodalization). To allow for 1-D modeling approximation of flow in the core, the diverging and converging regions (truncated cones) of the core are modeled as equivalent cylindrical regions. The effective diameters of these cylinders are calculated such that the volumes, thus the mass and stored energy, of the converging or diverging regions are preserved. For preserving the heat transfer through the walls of the converging or diverging regions, separate effective heated perimeters are calculated and used only for that purpose. These assumptions are adequate for simulation of the KP-FHR test reactor for postulated event analysis provided that the biases and assumptions required to perform conservative safety analysis, such as the HPFs, are appropriately applied. All other fluid (Flibe) flow regions (e.g., cold leg, downcomer, hot leg, inlet flow channel, rim gap) are also modeled in 1-D, with nodalization only along the direction of the flow. Heat conduction in solid structures is modeled in 2-D.

The pebble heat structures in the reactor are modeled and nodalized in spherical 1-D as follows:

- the fuel pebbles are modeled as a three region heat structure consisting of the pebble core, the fueled core, and the pebble shell;
- the moderator pebbles are modeled as a single region heat structure; and
- the TRISO fuel particles are modeled using the KP-SAM TRISO particle model.

This nodalization and modeling is adequate for the purposes of the safety analysis methodology, provided that the output temperatures of the simulation are used as an input to the KP-BISON code for further analysis. The staff notes that the KP-SAM calculations use conservative hard-coded material properties, which reduce the calculated margin to fuel safety limits, and the final calculation of the fuel safety limits are performed using KP-BISON which explicitly captures changes in thermal properties as a function of temperature and irradiation.

The graphite reflector is composed of stacked blocks which lead to the formation of a series of both horizontally-oriented and vertically-oriented interstitial radial gaps between the reflector blocks during reactor operation, as well as a rim gap between the reflector and the core barrel to accommodate differential thermal expansion. The reflector blocks also include flow channels to accommodate instrumentation and control equipment. The paths through the pebble bed and through the annular rim gap are explicitly modeled in KP-SAM, but the paths through the gaps in the reflector are not modeled due to the uncertainty associated with their detailed geometry.

This graphite reflector configuration introduces uncertainty in modeling the bypass and core flow distribution, especially given the lack of experimental data and the use of 1-D modeling and uniform porosity in the pebble bed region. The CFD **[[ ]]** model from KP-TR-024-NP can provide some insights into the steady state bypass and core flow distribution. However, there is a lack of understanding of how that flow would develop during the initial transient phase and what its impact would be on the temperature distribution. Although the impact of uncertainty associated with dynamic flow distribution during transients may be reduced by conservative flow distribution assumptions based on each transient and by assumptions defining conservative surrogates for FOM temperatures, this is an area where the final modeling approach based on final design, sensitivity analysis results, and any applicable validation data, must be further reviewed and approved. Therefore, the staff is imposing **Limitation and Condition 4 (c)** to ensure that an application referencing this TR provides for review the justifications (e.g., bypass flow sensitivity analyses) that the transient bypass and core flow distribution as modeled results in conservative prediction of the temperature distribution.

For the salt spill event analysis, valve components in KP-SAM are used to model the anti-siphon features on the PSP and the cold leg of the PHTS. A valve component, which is added directly below the PSP component, closes and stops the flow into the PSP when the Flibe level in the upper plenum is below the axial location of the PSP anti-siphon level. A valve component, which is added on the cold leg component, closes and stops the flow into the cold leg when the Flibe level in the upper plenum is below the cold leg anti-siphon level. The pipe break in the PHTS piping is modeled using three additional valves as described in TR section 4.3.10. The staff finds this modeling approach for salt spill event analysis to be acceptable because the modeling approach provides an adequate representation of a salt spill event and system response.

In general, a typical element or node size of approximately 10 cm, with a minimum of three elements, is used in developing the mesh, which is typical and appropriate in most regions. In regions with steep gradients, a finer mesh is used to appropriately maintain mass and energy balances and to ensure that the solutions variables are appropriately captured. As stated in the TR, the adequacy of the model nodalization scheme is assessed as part of the overall EM assessment, which includes dedicated mesh sensitivity studies to confirm independence of the simulation solutions from spatial discretization. The staff did not review the specific mesh sensitivity studies for the base model since the model does not represent the final design, and the staff was not asked to review it. The staff will review these sensitivity studies for the final EMs when they are submitted. According to limitation 3 in TR section 6.2, the base model will be updated using the validation insights. This would also include confirmation of the adequacy of the nodalization selected for the base model. Specifically, the IET validation is expected to play a key role in confirming the 1-D nodalization approach selected for modeling the core. The staff includes this limitation through **Limitation and Condition 1**.

The staff notes that this SE does not approve a specific KP-SAM input model. The SE approval is limited to the general approach for the base model development. Review of the final base model and its adequacy to represent the final geometry and the initial and boundary conditions will be addressed in a future licensing submittal.

TR sections 4.1.1 and 4.3.2 discuss the approach used to calculate the power response of the reactor core during transients simulated by KP-SAM. The fission power is calculated via KP-SAM's point kinetics approximation, accounting for changes in reactor power due to prompt and delayed neutron generation. Reactivity feedback behavior is included in the model via core-level coefficients generated with the applicant's nuclear analysis methodology described in KP-TR-024-NP. Temperature feedback coefficients are included for the fuel, moderator pebbles, coolant and reflector. **[[**

]] In addition to feedback coefficients, kinetics parameters (e.g., delayed neutron group fractions and lifetimes, mean generation time) and RSS trip worth are provided by nuclear design. Nuclear design calculations are also used to develop reactivity worth curves for inadvertent RCS withdrawal. Uncertainties in all these parameters are accounted for via the nuclear reliability factors (NRFs) developed as part of nuclear analysis and are biased according to the NRFs (and additional margin, if needed) on an event-specific basis. RSS trip worth is also penalized via maintaining the highest-worth control blade stuck in a fully withdrawn position. The axial power distribution used in generating the reactor trip worth is biased relative to nominal towards the bottom of the core to delay the negative reactivity of the blades entering the active core region. The power distribution used within KP-SAM simulations are biased relative to nominal towards the top of the core to penalize natural circulation flow and therefore create more challenging resultant temperature conditions. The staff finds the applicant's implementation of point kinetics to represent the reactor core power transient to be acceptable because it (1) is an implementation of a common safety analysis approach; (2) considers all relevant feedback mechanisms and external reactivity sources; and (3) appropriately considers uncertainty to produce conservative evaluations of the core and system response to an upset condition.

TR section 4.1.1 also describes that decay heat can be modeled in KP-SAM via a user-supplied decay power vs. time curve or using an internal predictive model. For an internal predictive model, core design calculations would be utilized to determine the relative fraction of fissions coming from uranium-235, uranium-238, plutonium-239 and plutonium-241 based on the core composition at the desired time in life at which the transient would be simulated. The predictive model can be further manipulated via a multiplicative factor to introduce additional conservatism in decay heat calculations as desired. The staff confirmed during its audit of Kairos' supporting documentation that the base Hermes KP-SAM model uses the nominal default decay heat model in KP-SAM, which is the same as that included in the base SAM code developed by Argonne National Laboratory. Review of the SAM theory manual (Reference 7) indicates that the default decay heat model is based on the American National Standard (ANS) standard ANSI/ANS-5.1-2005, "American National Standard Decay Heat Power in Light Water Reactors," (Reference 8). Although KP-FHRs are not light water reactors, it may be that this decay heat model is adequate for simulation purposes. However, justification for this or any other decay heat model is not included in the TR. The staff finds that the use of a decay heat model is acceptable and necessary for adequate evaluation of a KP-FHR transient. However, because no specific model has been justified at this time, the staff is imposing **Limitation and Condition 4 (d)**, that requires an applicant utilizing this safety analysis methodology to justify and submit the specific decay heat model and associated uncertainties incorporated in licensing submittals to the staff for review.

4.3 KP-SAM Validations and Applicability Evaluation

EMDAP element 2 provides guidance on development of an assessment base for the EM. The assessment base generally includes analytical (or fundamental) assessments, SET assessments, and IET assessments. The SETs are used to validate empirical correlations and closure models used in the EM (e.g., **[[ ]]**). The IETs allow validation of the EM for prediction of systems interactions and global phenomena (e.g., **[[ ]]**). The purpose of the analytical or fundamental validations is to illustrate fundamental calculational capability of the EM (e.g., conduction heat transfer). TR section 4.2 describes the analytical validations (verification), SETs, and IETs selected for the validation of the KP-SAM EM.

Table 3-1 of the TR presents a summary of the thermal fluids PIRT and describes the treatment of high and medium ranked phenomena in the KP-SAM EM. Some of the PIRT phenomena are modeled using first principal models (e.g., **[[ ]]**). The validation of KP-SAM for modeling such phenomena is performed using analytical simulations or numerical verifications and is described in TR section 4.2.1. Some phenomena in the PIRT are either conservatively neglected or modeled using a conservative bounding approach in the KP-SAM EM; therefore, the staff finds that no specific validations are needed for these types of phenomena. Examples of these phenomena include **[[ ]]**.

TR section 4.2.2 describes the SETs planned for the validation of key empirical closing relations used in the KP-SAM model. These closing relations are summarized in SE section 4.1.2 and include **[[ ]]**. TR section 4.2.2.1 describes the SET planned for the validation of **[[ ]]** TR section 4.2.2.2 describes the planned SET for **[[ ]]**.

TR section 4.2.3 describes the only IET planned for the KP-FHR test reactor. This test is planned to **[[ ]]**. The TR states that the top-down and bottom-up scaling analysis will be performed in accordance with TR KP-TR-006-NP-A, “Scaling Methodology for the Kairos Power Testing Program,” (ML21013A430), to ensure the applicability of the IET data to the KP-FHR test reactor.

The staff finds that the TR provides an adequate description of the KP-SAM validation database consistent with EMDAP step 5. Furthermore, the scaling methodology, previously reviewed and approved by the staff in KP-TR-006-NP-A satisfies EMDAP step 6. However, EMDAP steps 7 through 9 are not addressed in the TR for the planned SETs and IET data. Limitation 3 in TR section 6.2 clarifies that the validation and code assessment results will be provided for the phenomena described in TR section 4.2. However, as indicated by EMDAP steps 7 through 9, the application of scaling methodology to evaluate the scalability of SETs and the distortions in

IET data is essential. Any limitations identified by the scalability and distortion analysis must be addressed, consistent with the applicability evaluation under EMDAP Element 4. Therefore, the staff is imposing **Limitation and Condition 4 (e)** that ensures an application referencing this TR provide for staff review scalability and distortion analysis for the SET and IET data and justifications for any limitations identified by this analysis through the applicability evaluation under EMDAP Element 4.

As discussed in SE section 3.3, **[[**

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TR section 4.3.5 states that the DHRS in KP-SAM is modeled using the heat flux boundary conditions that are tuned to match the DHRS performance curve. Furthermore, the TR states that the DHRS performance curves are set to provide best-estimate, minimum, or maximum heat removal to bound the DHRS heat transfer rates and will be verified through testing. The TR does not provide any further information on the planned DHRS testing and the derivation of the bounding DHRS performance curves. As described in SE section 3.1, DHRS is a critical safety system that plays a key role in mitigation of many postulated events in the KP-FHR test reactor. Therefore, the staff is imposing **Limitation and Condition 4 (f)** that ensures an application referencing this TR include for staff review: (1) the DHRS test data characterizing DHRS performance and phenomena under steady-state and transient conditions used to confirm the DHRS performance curves and the applicability evaluation that describes the scaling and distortion analysis of the DHRS tests; and (2) how the DHRS performance curves bound the identified distortions.

4.4 KP-SAM Documentation, Quality Assurance and Configuration Control

The code description and supporting documentation for the safety analysis methodology described in the TR are consistent with KP-SAM code version **[[ ]]**, which is based on the SAM code developed by ANL. The SAM code was evaluated as appropriate for use as the basis for KP-SAM version **[[ ]]** through a commercial grade dedication process in accordance with the Kairos Power software quality assurance program. The KP-SAM code is maintained and its activities managed (i.e., configuration control) under the Kairos Power software quality assurance program. TR section 4.2 describes the software verification process for KP-SAM that includes verification using fast regression tests and numerical verification using analytical tests for simple models such as the fluid models, heat conduction, numerical methods, component models (e.g., pebble bed core channel, reactor power, tank, and heat structure components), and reactor core physics (e.g., decay heat and point kinetics models). The TR provided the following documentation:

- EM requirements documented in section 3 of the TR including the PIRTs
- EM documented in this TR and in other interfacing TRs including:
  - KP-TR-010-P-A
  - KP-TR-024-NP
- Code description manuals provided for SAM and KP-SAM:
  - SAM theory manual, ANL/NSE-17/4
  - KP-SAM theory manual, KP-RPT-000231
- Code user manuals and guidelines provided for SAM and KP-SAM:
  - SAM user’s guide, ANL/NSE-19/18 (Reference 9)
  - KP-SAM users guide, KP-RPT-000226

- Scaling report, KP-TR-006-NP-A
- Assessment reports include numerical verifications against the analytical solutions described in TR section 4.2.1 and planned validation of KP-SAM against SET and IET data as described in TR sections 4.2.2 and 4.2.3.

The staff finds that the TR provides adequate information on the EM documentation, quality assurance, and configuration control because it is consistent with the types of information indicated by the guidance for implementing EMDAP step 10.

5. Event-Specific Methods

Section 5 of the TR describes the biases applied and sensitivity studies to be conducted for each event category. The event-specific biases and sensitivities account for uncertainties and are used to identify the limiting events for each event category consistent with guidance in NUREG-1537, part 1, chapter 13. The event-specific methods consider analytical limits, core design inputs, reactor physics parameters, coolant properties, treatment of non-safety and auxiliary SSCs, control systems, and operator actions. TR section 5 also describes the event-specific source-term analysis methods for the salt spill and PHSS malfunction event categories.

5.1 Event-Specific Biases and Sensitivities for KP-SAM Model

The biases for the following parameters are applied to the KP-SAM base model for analyzing the postulated event categories with an intact primary side (i.e., increase or decrease in heat removal, loss of forced circulation, and reactivity-initiated event) and the salt spill postulated event group:

- Reactor initial power
- Reactivity coefficients (Doppler, moderator, coolant, reflector)
- Reactor kinetics parameters (neutron mean generation time, effective neutron fractions, delayed neutron precursor decay constants)
- Power distribution
- Shutdown margin and shutdown element insertion time
- DHRS capacity
- Decay heat
- Heat rejection blower performance
- RPS actuation delays (instrument response time and actuation delays)
- Reactor coolant thermophysical properties and material properties

Additional biases considered for the salt spill event category include:

- System pressure (biased high to maximize break flow rate)
- Anti-siphon activation level (biased low to delay the onset of natural circulation cooling)
- Inventory management system recirculation flow and Flibe volume (biased high to delay the actuation of the RPS level trip)

The sensitivity calculations are proposed for KP-SAM model calculations to identify the limiting event in each event category. These calculations also include biases discussed earlier. The parameters considered for the sensitivity calculations include:

- Reactor initial power
- Reactor coolant average temperature
- Material properties
- Heat rejection radiator (HRR) outlet temperature
- Plant control systems (reactivity control and primary salt pump control)
- Bypass flow fraction
- Operation of PSP and HRR blower

In addition, for the reactivity-initiated event group, sensitivities on the magnitude of reactivity insertion are performed to simulate the range on control element withdrawals. The limiting event is defined as the scenario(s) that results in the least amount of margin to the acceptance criteria for the FOMs. The staff finds that the biases and sensitivities proposed for the KP-SAM model are appropriate for the identification of limiting events and for the conservative estimation of the safety analysis methodology FOMs including peak TRISO SiC temperature, fuel failure fraction, peak vessel temperature, peak reflector temperature, and energy deposition pulse width because they address key uncertainties in the safety analysis methodology.

TR sections 5.5.1 and 5.5.2 also list assumptions and biases used to analyze the event-specific release pathways using the source term method for the salt spill and PHSS malfunction events, respectively. The evaluation of source term methods proposed for the calculations of dose for these event categories is provided in SE section 5.2 below.

5.2 Event-specific Source Term Methods

Certain postulated event scenarios include radionuclide transport mechanisms or pathways that are not included in the MHA or an intact primary system event. These events are the salt spill, the PHSS malfunction, and radioactive release from subsystem or component. TR sections 5.5.3 and 5.6.2 provide source term methods for these additional transport mechanisms for the salt spill events and PHSS malfunction events, respectively. The staff's understanding of the event-specific source term methods was aided by the example calculations A.3 (salt spill) and A.4 (PHSS malfunction) provided in appendix A of the TR. Appendix A is not part of the safety analysis methodology and therefore the staff did not review appendix A and the specific assumptions used therein for acceptability.

Salt Spill Events

As stated in TR section 5.5, for the salt spill events, the event-specific radionuclide release pathways that are additional to those modeled in the MHA are:

- Single-phase and two-phase mechanical aerosol generation through the break
- Splashing of Flibe onto the catch pan
- Release of radionuclides from the spilled Flibe pool via evaporation
- Tritium and Ar-41 releases from oxidized graphite in the reflector

The salt spill event EM uses the approved methods in the MST methodology to evaluate the single-phase mechanical aerosol generation through the break and the aerosols generated through splashing of Flibe on the catch pan, as well as the gas space transport. TR section 5.5.3 provides methods for two-phase aerosol generation, evaporative release from the spilled Flibe pool, and releases from the oxidized graphite.

For the salt spill postulated event group, the entire spectrum of break sizes and locations must be considered to identify the limiting conditions for each release pathway. The limiting event is modeled as a combination of the most limiting break conditions rather than a single break scenario. The TR describes a method using weighted release fractions for each release pathway. The TR acknowledges that the limiting break size may be different for mechanical aerosol generation than for splashing. The staff finds that this approach is acceptable because it appropriately maximizes the potential radiological releases through aerosol generation and splashing.

As described in TR section 5.5.3, the initial Flibe release through a break is modeled as single-phase flow while the Flibe-cover gas interface is above the siphon break point. After the interface falls below the siphon break point, the release is modeled as two-phase flow, which results in a change in the mechanical aerosol generation. The salt spill EM uses the aerosol release modeling for single-phase jet breakup from the approved MST methodology. The model for mechanical aerosol generation from two-phase flow uses an empirical correlation for a break in the side of a pressurized tank to determine the quality of the two-phased break flow. The Sauter Mean Diameter (SMD) is a representative diameter for the distribution of aerosol particle sizes. The method derives the SMD for two-phase flow based on an empirical correlation from a study for a liquid jet injected into a high-velocity gas stream. Then, the SMD is used in the Rosin-Rammler particle size distribution formula (described in reference 36 of the TR) to calculate the airborne release fraction from the two-phase flow jet breakup. The staff evaluated the information in TR section 5.5.3 and finds that the assumptions are appropriate because they are reasonably based on theoretical correlations to physical phenomena, use widely accepted aerosol modeling concepts such as SMD and the Rosin-Rammler distribution, and the supporting references are relevant to the modeling of aerosol formation from molten salt two-phase break flow.

TR section 5.5.3 states that the same evaporation model used for in-vessel releases in the MHA is used to calculate evaporative releases from the spilled Flibe pool. The pool surface temperature curve is calculated by the KP-SAM pool cooling model. The TR states that use of the in-vessel release model is conservative for releases from the spilled Flibe pool even though spilled Flibe could potentially form more volatile compounds through interaction with oxygen than are formed on the Flibe surface in the intact primary system such as for the MHA. This is compensated for by the model neglecting the formation of a crust on a shallow pool surface which would prevent evaporative releases. For deeper pools, the TR provides a basis that the relatively small size of the liquid-vapor interface compared to the bulk spilled Flibe would reduce the potential for interaction with oxygen and that releases are limited as the Flibe cools. The staff finds that the use of the MHA in-vessel evaporative release model is reasonably supported by the TR discussion and is acceptable for the spilled Flibe because it is based on the approved MST methodology in KP-TR-012 and subject to the limitations and conditions on the Flibe release modeling in the approval of KP-TR-012.

The salt spill event evaluation model uses the same methods as the MHA to determine the tritium and Ar-41 MAR in the graphite reflector material. The model assumes a full and instantaneous release of MAR from the exposed reflector graphite. The staff finds this modeling to be conservative and, therefore, is acceptable.

PHSS Malfunction Events

The PHSS malfunction event group source term is based on a scenario biased to ensure that the maximum amount of MAR is released. The PHSS malfunction event EM described in TR section 5.6.2 provides source term methods for the following release pathways not modeled in the MHA:

- Releases from exposed TRISO kernels via diffusion
- Releases from dissolved TRISO particles in Flibe via evaporation and off-gassing
- Releases from graphite dust resuspension
- Tritium and Ar-41 releases from oxidized graphite in the reflector and pebble matrix material

For releases from the oxidized pebbles, the MAR in the small fraction of exposed TRISO kernels (i.e., with a failed SiC layer pre-transient) is released through diffusion driven by an assumed gas-space temperature of 650°C, which is 100°C hotter than the nominal temperature. This allows for modeling of diffusion from the pebbles without any potential retention in Flibe. Radionuclide transport modeling from the fuel and gas space is taken from the approved MST methodology and is consistent with the modeling in the MHA. The remaining TRISO particles (i.e., with an intact SiC layer) in the oxidized pebbles are assumed to fall and be dissolved into the Flibe as a conservative assumption. The SiC layer in these particles is assumed to fail instantaneously with release of MAR into the Flibe. The time-temperature curve for the hot well is used in the modeling of evaporative release from the Flibe. The release from the Flibe is modeled as evaporative releases due to prolonged temperature conditions using the approved MST methodology in KP-TR-012. The staff finds the assumptions on radioactive transport and release from the TRISO particles and Flibe to be conservative and based on previously approved methods, and, therefore, are acceptable.

TR section 5.6.1 describes the method to estimate the amount of activated graphite dust accumulated in the pebble transfer line. For the source term modeling of releases from graphite dust resuspension, the MAR in the graphite dust is assumed to puff release to the gas space at the initiation of the transient. The staff finds this modeling to result in a conservative estimation of the dose consequences from release of graphite dust and, therefore, is acceptable.

The PHSS malfunction event evaluation model uses the same methods as the MHA to determine the tritium and Ar-41 MAR in the graphite reflector and pebble matrix material prior to the transient. The model assumes a full and instantaneous release of MAR from the exposed reflector graphite and oxidized pebbles. Furthermore, the model uses the approved methods in the MST methodology in KP-TR-012 to evaluate the gas space transport. The staff finds this modeling to be acceptable because it is consistent with approved methods and the assumptions would result in a conservative estimation of the dose consequences of the release from the exposed graphite and oxidized pebbles.

Release from Subsystem or Component Events

TR section 5.7 describes the source term modeling for the radioactive release from subsystem or component events. To simplify the dose calculations for the FOM comparison, each source of MAR in the subsystem or component is modeled as an activity quantity of a representative “effective” isotope for which the amount of the isotope alone would produce the same TEDE as the combined activities of the isotopes in the MAR. The MAR is assumed to puff release at the initiation of the transient without modeling of holdup and retention in the building, radioactive decay, or delayed releases, which leads to a conservative estimate of TEDE at the EAB. The approved MST methodology in KP-TR-012 is used to determine the MAR in the subsystem or component and modeling of transport in the gas space and building. Therefore, the staff finds that the source term modeling described for the radioactive release from subsystem or component events is acceptable because it is consistent with approved methods and the assumptions would result in a conservative estimation of the dose consequences.

LIMITATIONS AND CONDITIONS

Kairos included six limitations on the use of this TR in section 6.2 which the staff includes through **Limitation and Condition 1**. The staff considered these limitations in the technical review presented in this SE. The staff applies the following additional limitations and conditions on the acceptance of this TR:

1. A test reactor OL application referencing this TR must demonstrate that the limitations listed in TR section 6.2 are met, subject to staff review and approval.
2. The safety analysis methodology for modeling a PHSS malfunction event is limited to modeling only events initiated by a break in the transfer line. The safety analysis methodology is not applicable to simulate any other PHSS malfunction event. A test reactor OL application referencing this TR should identify the limiting event for the PHSS malfunction event category and provide an applicable methodology for the limiting event.
3. The KP-BISON methodology for calculation of peak TRISO SiC layer temperature and fuel failure fraction FOMs is not reviewed and approved in this SE. A test reactor OL application referencing this TR must provide for staff review acceptable justification for use of KP-BISON for the calculation of these FOMs.
4. A test reactor OL application referencing this TR must provide for staff review:
    a. acceptable material qualification data that supports the acceptance criteria for the peak vessel temperature and peak structural graphite temperature FOMs;
    b. 2-D/3-D static core analysis and the CFD **[[ ]]** model results used for the calculation of peaking factors and local flow penalty factor in the HPF and the reflector temperature surrogate factor;
    c. justifications that the transient bypass and core flow distribution as modeled results in conservative prediction of the temperature distribution;
    d. specific decay heat model and relevant uncertainties;
    e. scalability and distortion analysis for the SET and IET data and justifications for any limitations identified by this analysis through the applicability evaluation under EMDAP Element 4; and
    f. DHRS test data characterizing DHRS performance and phenomena under steady-state and transient conditions used to confirm the DHRS performance curves and the applicability evaluation that describes the scaling and distortion analysis of the DHRS tests and how the DHRS performance curves bound the identified distortions.

CONCLUSION

The staff concludes that Kairos's KP-TR-020-P, "Safety Analysis Methodology for the Kairos Power Fluoride Salt-Cooled High-Temperature Test Reactor," Revision 1, provides an acceptable safety analysis methodology for evaluating the MHA and postulated event dose consequences for the KP-FHR test reactor subject to the limitations and conditions discussed above. The evaluation of final compliance or conformance with the identified regulations and PDC will be performed during the review of a licensing application referencing this TR.

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Principal Contributor(s): Pravin Sawant
Michelle Hart
Tarek Zaki
Andrew Bielen
Ben Adams
Alex Siwy
Alexander Chereskin

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