

KP-NRC-2512-001

Enclosure 2

**Approved Version of KP-FHR Core Design and Analysis Methodology Topical Report
(Non-Proprietary)**

Section A

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A	Letter from Joshua Borromeo to Darrell Gardner, “Kairos Power LLC - Final Safety Evaluation of Topical Report, KP-FHR Core Design and Analysis Methodology, Revision 1,” November 25, 2025 (Non-Proprietary)
B	Kairos Power Topical Report: “KP-FHR Core Design and Analysis Methodology”, KP-TR-024-NP-A, Revision 1 (Non-Proprietary)



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

November 25, 2025

Darrell Gardner
Sr. Director, Licensing
Kairos Power LLC
707 W Tower Ave
Alameda, CA 94501

SUBJECT: KAIROS POWER LLC – FINAL SAFETY EVALUATION OF TOPICAL REPORT,
“KP-FHR CORE DESIGN AND ANALYSIS METHODOLOGY,” REVISION 1
(EPID NO: L-2024-TOP-0013)

Dear Darrell Gardner:

This letter provides the final safety evaluation (SE) for the Kairos Power, LLC (Kairos) topical report (TR) “KP-FHR Core Design and Analysis Methodology,” Revision 1. By letter dated April 3, 2024, Kairos submitted for U.S. Nuclear Regulatory Commission (NRC) staff review KP-TR-024-P, “KP-FHR Core Design and Analysis Methodology,” Revision 0 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML24095A255). On August 22, 2024, the NRC issued an audit plan for the purpose of gaining a better understanding of KP-TR-024-P (ML24222A276). By letter dated June 17, 2025, Kairos submitted Revision 1 of KP-TR-024-P (ML25168A340). The NRC staff issued the audit summary report on November 20, 2025 (ML25142A086).

The NRC staff’s final SE for KP-TR-024-P, “KP-FHR Core Design and Analysis Methodology,” Revision 1 is enclosed. The NRC staff provided Kairos with a draft of the SE for the purpose of identifying proprietary export-controlled information on July 25, 2025 (ML25206A085). On November 21, 2025, Kairos confirmed that the proprietary export-controlled information in the final SE was appropriately marked.


The NRC staff requests that Kairos publish an accepted version of this TR within 3 months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed SE after the title page. The accepted version shall include an “-A” (designated accepted) following the TR identification number.

The Enclosure to this letter contains Proprietary
Export Controlled Information. When separated from
the Enclosure this letter is DECONTROLLED

If you have any questions regarding this matter, please contact Cayetano Santos Jr. via email at Cayetano.Santos@nrc.gov.

Sincerely,

**JOSHUA
BORROMEIO**

 Digitally signed by JOSHUA
BORROMEIO
Date: 2025.11.25 13:19:49 -05'00'

Josh Borromeo, Chief
Advanced Reactor Licensing Branch 1
Division of Advanced Reactors and Non-Power
Production and Utilization Facilities
Office of Nuclear Reactor Regulation

Project No.: 99902069

Enclosure:
Safety Evaluation

cc: Kairos Power Hermes via GovDelivery

SUBJECT: KAIROS POWER LLC – FINAL SAFETY EVALUATION OF TOPICAL REPORT,
“KP-FHR CORE DESIGN AND ANALYSIS METHODOLOGY,” REVISION 1
(EPID NO: L-2024-TOP-0013) DATE: NOVEMBER 21, 2025

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**KAIROS POWER LLC – FINAL SAFETY EVALUATION OF TOPICAL REPORT
KP-TR-024-P, “KP-FHR CORE DESIGN AND ANALYSIS METHODOLOGY”
(EPID L-2024-TOP-0013)**

SPONSOR AND SUBMITTAL INFORMATION

Sponsor: Kairos Power LLC (Kairos)

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Project No.: 99902069

Submittal Date: April 3, 2024

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Revision Letter Date and ADAMS Accession No: Revision 1, June 17, 2025, (ML25168A340)

Brief Description of the Topical Report: On April 3, 2024, Kairos Power LLC (Kairos) submitted Topical Report (TR) KP-TR-024-P, “KP-FHR Core Design and Analysis Methodology,” Revision 0 (Agencywide Documents Access and Management System (ADAMS) Accession No.:ML24095A255) for the U.S. Nuclear Regulatory Commission (NRC) staff review. On June 17, 2025, Kairos submitted Revision 1 of this TR (ML25168A340). The TR provides the methodology for core physics, thermal-hydraulic analysis, and radiation effects on materials for the Kairos Power Fluoride Salt-Cooled, High Temperature Reactor (KP-FHR). The methodology described in the TR is used to calculate reactivity coefficients, control and shutdown rod worths, shutdown margin, flux distribution, power distribution, temperature distribution, kinetics parameters, material depletion, radiation damage and heating, and pebble peaking factor. The TR identifies the computer codes used and discusses their verification and validation. The TR also describes an approach for quantifying uncertainties and determining biases to inform the development of nuclear reliability factors (NRFs).

EVALUATION CRITERIA

Regulatory Requirements

Title 10 of the *Code of Federal Regulations* (10 CFR) 50.34(a)(4) for a construction permit (CP) application, 10 CFR 50.34(b)(4) for an operating license (OL) application, and 10 CFR 52.79(a)(5) for a combined license (COL) application require, in part, analysis and evaluation of the design and performance of structures, systems, and components (SSCs) of the facility with the objective of assessing the risk to public health and safety resulting from

Enclosure

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 core design and analysis methodology presented in the TR is used to determine safety margin for the KP-FHR fuel and core during normal operation. The methodology also provides input to the safety analysis performed to determine adequacy of SSCs designed to prevent or mitigate the postulated events in a KP-FHR.

Topical report KP-TR-003-NP-A, "Principal Design Criteria for the Kairos Power Fluoride Salt-Cooled, High Temperature Reactor," Revision 1, dated June 12, 2020, (ML20167A174) provides principal design criteria (PDC) for the KP-FHR design that were reviewed and approved by the NRC staff. Section 1.2 of the core design and analysis methodology TR identifies PDC that can be addressed, in part, using the methodology presented in the TR. These include PDC 10, 11, 12, 16, 25, 26, 28, 31, and 34.

While the NRC staff considered these regulations and PDC in its review of the TR, determinations regarding compliance or conformance will be made during review of licensing applications referencing this TR.

TECHNICAL EVALUATION

The Kairos core design and analysis methodology presented in the TR is based on the Serpent 2 computer code for the nuclear design and the STAR-CCM+ computer code for the thermal-hydraulic analysis. The methodology models steady-state neutronic and thermal-hydraulic phenomena and is developed for startup, power ascension, and equilibrium conditions in KP-FHR test and power reactors. As described in TR section 1, "Introduction," Kairos is requesting the NRC staff's review and approval of the following:

- use of the Serpent 2 and STAR-CCM+ based calculational framework to calculate reactivity coefficients, control and shutdown rod worths, kinetics parameters, power distribution, material depletion, shutdown margin, flux distribution, temperature distribution, radiation damage and heating, and pebble peaking factor
- methodology for quantification of biases and uncertainties in neutronic calculations based on code-to-code benchmarks and literature data
- methodology for validation of thermal-hydraulic models and quantification of biases and uncertainties in thermal-hydraulic calculations
- methodology for updating NRFs using operational data

1. Topical Report Overview

The TR consists of the following major sections and appendices:

Section 1, "Introduction," provides a brief description of KP-FHR design features and identifies specific technical areas for which Kairos is requesting NRC staff review and approval. The NRC staff considers the information in section 1 of the TR throughout the technical evaluation in this safety evaluation (SE) but does not make any determinations on the information in TR section 1.

Section 2, “KP-FHR Core Design Features,” describes the KP-FHR core design and introduces KP-FHR operational regimes considered in the TR. Limitation 1 in TR section 7.2, “Limitations,” states that the methodology is applicable to the design as presented in this TR; the NRC staff includes this limitation, and other limitations cited in TR section 7.2, through **Limitation and Condition 1** in the “Limitations and Conditions” section of this SE. The NRC staff considers the information in section 2 of the TR throughout the technical evaluation in this SE but does not make any determination on the information in TR section 2.

Section 3, “Core Modeling Paradigms,” describes the modeling approach for the KP-FHR core, which includes the discrete element method (DEM) for modeling pebble movement, a neutronic model for modeling neutron transport, and a thermal-hydraulic model for modeling coolant flow and temperature distribution. It also summarizes the key steady-state phenomena used to determine the adequacy of the methodology to calculate important figures of merit (FOMs) and provides a list of parameters or outputs calculated by the core methodology.

Section 4, “Modeling Tools,” provides a summary of the computer codes, STAR-CCM+ and Serpent 2, and several supporting computer programs or tools used to implement the core methodology models. The NRC staff’s evaluation of the KP-FHR core methodology phenomena, modeling approach, associated computer tools, and adequacy to calculate the parameters in TR section 3.6 is found in SE section 2, “Methodology Approach.”

Section 5, “Validation, Verification, and Uncertainty Analysis,” discusses the validation, verification, and uncertainty quantification of the DEM, neutronics, and thermal-hydraulic core models. This section discusses use of computational fluid dynamics (CFD) for numerical validation of the thermal-hydraulic model. It also discusses startup testing and the methodology for calculating and updating NRFs. The NRC staff’s review of the validation, verification, and uncertainty analysis approach is found in SE section 3, “Methodology Validation, Verification, and Uncertainty Analysis.” TR appendix B, “Verification and Validation,” provides additional discussion on validation of the DEM and thermal-hydraulic models. The NRC staff used the information in TR appendix B to enhance its understanding of the methodology but does not make any determinations on TR appendix B.

Section 6, “Applications,” discusses applications of the methodology, including how FOMs and certain outputs are used for safety analysis, source term, nuclear design, and thermal-hydraulic analysis. It also discusses startup physics testing and the methodology to calculate the desired core composition. TR appendix A, “Example Calculation,” illustrates the application of the methodology. The NRC staff considered the information in section 6 and appendix A of the TR throughout the technical evaluation in this SE but does not make any determinations on the information in TR section 6 and appendix A.

Appendix C, “Neutronics PIRT Results for the KP-FHR,” provides a list of neutronic phenomena ranked by importance and knowledge level. The NRC staff considered the information in the phenomena identification and ranking table (PIRT) in appendix C for evaluation of the neutronic model in SE section 2 but does not make any determinations on the PIRT in TR appendix C.

2. Methodology Approach

2.1 Methodology Overview and Computer Codes

The Kairos core design and analysis methodology consists of three models (paradigms): DEM, thermal hydraulics, and neutronics. The modeling domain for each model is shown in figure 3-1,

“Core Modeling Domains,” of the TR. Table 1, “Methodology Models and Codes,” of this SE summarizes these models. Table 2, “Computer Codes Used,” of this SE summarizes the computer codes used to implement the models and the wrapper tools that facilitate the coupling and/or data transfer between the models.

DEM calculates the movement of pebbles in the inlet fueling region, cylindrical and conical (converging and diverging) core regions, and defueling chute. The methodology uses a commercially available CFD code, STAR-CCM+, to implement DEM. DEM calculates static pebble center locations, packing fractions (bed porosity), and average pebble tracks and velocity profiles that are used in the neutronic and thermal-hydraulic models. The DEM results are post-processed to generate spectral zones in the core for the neutronic calculations. DEM is coupled with the neutronic model using a wrapper code, KPACS. As described in TR section 4.4.3, “KPACS,” KPACS can simulate pseudo-steady-state evolution of the KP-FHR core. The NRC staff’s evaluation of the DEM model is provided in SE section 2.2, “Discrete Element Method (DEM).”

The neutronic model calculations are performed using Serpent 2, a continuous-energy Monte Carlo (MC) neutron transport code. As shown in TR figure 3-1, the modeling domain for the neutronic model includes a full three-dimensional (3-D) representation of the core (including converging and diverging sections), defueling chute, fueling region, reactor shutdown system (RSS), reactor control system (RCS), graphite reflector, reactor penetrations, core barrel, downcomer, and reactor vessel. The core model explicitly represents different types of pebbles and tri-structural isotropic (TRISO) particles. The reflector is modeled with major engineered penetrations and geometries for the coolant paths. The neutronic model is explicitly coupled with the thermal-hydraulic model using the KPATH wrapper code. The power distribution calculated by the neutronic model is provided as input to the thermal-hydraulic model. The material temperature distributions calculated by the thermal-hydraulic model are used to update the material cross-sections in the neutronic model. The NRC staff’s evaluation of the neutronic model is provided in SE section 2.3, “Neutronics.”

The thermal-hydraulic model uses STAR-CCM+ to calculate the steady-state 3-D core material temperature distribution for the coolant (Flibe), fuel and graphite pebbles, TRISO particles, and reflector (TR section 7.2, Limitation 2). The model uses two paradigms: (1) a local thermal non-equilibrium (LTNE) porous media (PM) model for the core region, including fuel and moderator pebbles, and (2) a CFD [REDACTED] model for the reflector region, including the gaps and coolant flow channels. The NRC staff’s evaluation of the thermal-hydraulic model is provided in SE section 2.4, “Thermal-Hydraulic Model.”

In addition to KPACS and KPATH, TR section 4.4, “Wrapper Codes,” describes three more wrapper codes. Table 2 of this SE provides a brief description of the HEEDS, KACEGEN, and Zoner wrapper tools. TR section 4.4 states that these wrapper codes perform data transfer and do not contain physical models that need to be validated; however, these codes are numerically verified. TR figure 4-1, “High Level Process Flow Diagram of the Core Design and Analysis Methods,” illustrates a simplified process data flow from Serpent 2, STAR-CCM+, and wrapper codes to downstream applications. This figure illustrates the connections between computational modules, not inputs and outputs between computational modules. The NRC staff considered the information on these wrapper tools in the context of the overall calculational framework in SE section 2.5, “Overall Methodology Process.”

The TR also makes use of [REDACTED] in section 5.2, “Neutronics,” to perform code-to-code benchmarking and uncertainty evaluation.

Table 1 Methodology Models and Codes

Model	Code Used
DEM	STAR-CCM+
Neutronic Model	Serpent 2
Thermal-Hydraulic Model	STAR-CCM+

Table 2 Computer Codes Used

Computer Code	Description	Comments
STAR-CCM+	Multi-physics CFD code for thermal hydraulics and DEM calculations	[[[REDACTED]]] as described in TR section 3.5.1.
Serpent 2	Neutronic code used for a variety of calculations, including multiplication factor, control and shutdown element worths, reactivity coefficients, power distribution, kinetics parameters, nuclear heating, burnup, and activation analysis	Kairos made no modifications to the base Serpent 2 code.
HEEDS	Couples with STAR-CCM+ for input sensitivity and uncertainty analysis	Wrapper code (tool)
KACEGEN [Kairos ACE Generator]	Provides nuclear data for use in Serpent 2/MCNP6.2 by driving a nuclear data processing tool (NJOY21) to produce ACE-format nuclear data libraries	Wrapper code (tool)
KPACS	Couples Serpent 2 with DEM in STAR-CCM+; simulates pseudo-steady-state evolution of the KP-FHR core and [[[REDACTED]]]	Wrapper code (tool)
Zoner	[[[REDACTED]]]	Wrapper code (tool)

KPATH	Couples thermal-hydraulic model (STAR-CCM+) and neutronic model (Serpent 2).	Wrapper code (tool)
SCALE6.2	[[REDACTED]]	Neutronic code not evaluated in this SE
MCNP6.2	[[REDACTED]]	Neutronic code not evaluated in this SE

2.2 Discrete Element Method (DEM)

TR section 3.1.2, "Key DEM Phenomena," identifies core geometry and pebble contact, drag, and buoyancy forces as the important phenomena for the DEM model. These phenomena are modeled by a 3-D Lagrangian-Eulerian formulation as shown in equation 1 of the TR for the pebble momentum equation. Kairos clarified in audit discussions (ML25142A086) that the impacts of pebble geometry, coolant flow, and temperature (or density) distributions are accounted for through the modeling of forces on the pebbles.

The drag forces in DEM are calculated from [[REDACTED]] (see TR equation 2). [[REDACTED]] TR section 3.3 describes the [[REDACTED]] The NRC staff determined that this approach for calculation of drag forces is acceptable because the KP-FHR is characterized by a relatively slow moving, densely packed pebble bed in which [[REDACTED]] and the buoyancy force is pushing the pebble upward against the pebble retaining structure at the top of the core. The evaluation of [[REDACTED]] which is used to calculate [[REDACTED]] is discussed in SE section 2.4.1, "Local Thermal Non-Equilibrium (LTNE) Porous Media (PM) Model."

The buoyancy forces on the pebble are calculated by equation 3 of the TR and depend on pebble geometry and densities of coolant and pebbles. The NRC staff determined that this model for buoyancy force is acceptable because it is based on first principles.

The contact forces on pebbles are calculated by [[REDACTED]] shown by TR equation 4. It includes normal and tangential components. The NRC staff reviewed [[REDACTED]] implemented in STAR-CCM+ and notes that the tangential force component of the contact force is one of the most uncertain components in momentum balance due to [[REDACTED]] is presented in TR section 5.1, "Discrete Element Method (DEM)," and evaluated in SE section 3.1, "Discrete Element Method."

The NRC staff determined that the DEM modeling approach is acceptable because it is based on modeling of fundamental forces acting on the pebbles in a packed bed.

2.3 Neutronics

Serpent 2 is used for neutronic modeling of the reactor core and surrounding structures. It is a continuous-energy (CE) / constructive solid geometry (CSG) particle transport code that is capable of efficient, high-fidelity simulations of nuclear systems. As described in TR section 3.1.1, “Key Neutronics Phenomena,” the neutronic FOMs calculated by Serpent 2 within these workflows are the k-eigenvalue (i.e., k_{eff} or multiplication factor, which indicates system criticality) and power distribution, both of which are used as inputs to the safety analysis. Kairos used Serpent version 2.2.1 with no modifications for the analyses presented in this TR.

The NRC staff considered the neutronic PIRT presented in appendix C, “Neutronics PIRT Results for the KP-FHR,” and how phenomena are modeled in the neutronic methodology. Kairos used this internal PIRT, a previous PIRT performed at Georgia Institute of Technology for a similar technology, and literature review, to confirm that all necessary physics and phenomena were accounted for within Serpent 2. Kairos also used the PIRTs to help decide which neutronic parameters were sufficiently influential on the FOMs to warrant conservative biasing in safety-related calculations. Though PIRTs are typically used to guide evaluation model development, the NRC staff determined Kairos’s use of the PIRTs is reasonable because the result is still an evaluation model that accounts for key phenomena and processes in the KP-FHR. Furthermore, the sensitivity and uncertainty analysis methodology described in TR section 5.2.3, “Uncertainty Quantification,” provides additional information regarding which parameters warrant biasing.

TR section 3.4, “Neutronics Modeling Paradigm,” and TR section 4.3, “Serpent 2,” describe how the reactor neutronics are represented. CE cross-section libraries for Serpent 2 are prepared using the KACEGEN/NJOY21 process. An explicit randomized pebble- and TRISO-particle-resolved model of the pebble bed is generated via DEM simulation and input into Serpent 2. The pebble bed is placed within the graphite reflector structures that explicitly represent engineered cooling channels and major penetrations. The Flibe in minor flow regions (e.g., gaps between graphite blocks) is volume-homogenized into the graphite blocks. The RCS and RSS elements are represented with their absorber and cladding materials, neglecting other structural materials. RCS insertion is modeled by replacing Flibe within the associated channels in the reflector blocks with the RCS element to the desired depth. RSS insertion into the pebble bed is represented via the “cut pebble bed” method where regions of the pebble bed containing fuel pebbles and Flibe that overlap with RSS elements are replaced by the rods to the desired insertion depth. TR section A.1.3.3, “RSS Insertion Modeling Bias,” provides an [[

]] The Serpent 2 model includes all structures out to and including the reactor vessel, which allows for evaluation of dose and fluence to ex-core regions of the system.

Pebble depletion and the associated transmutation, activation, and decay of materials within the core is achieved using Serpent 2’s internal Bateman equation solver, which implements the Chebyshev Rational Approximation Method (CRAM). This method has become standard practice in depletion codes over the past decade.

The NRC staff reviewed the applicability of the Serpent 2 simulation software to model neutronics in the KP-FHR and determined it is acceptable because it appropriately captures all requisite physical features of the system and underlying physics. Assessments, verification of

the numerical techniques and modeling strategies, and applications of the Serpent 2 models are described in TR sections 5 and 6 and evaluated in SE section 3.2.

2.4 Thermal-Hydraulic Model

Kairos identified thermal-hydraulic phenomena important for modeling steady-state temperature distributions in the core (Flibe, pebble, TRISO) and reflector regions by performing sensitivity analyses on high-importance phenomena from the Kairos Power thermal fluid postulated event PIRT. For the core region, [[REDACTED]

]] were identified as important. For the reflector region, [[REDACTED]] were identified as important. During the audit Kairos clarified that the PIRT was not used to guide methodology development. It was primarily used to confirm that the methodology models important phenomena. [[REDACTED]

]] In addition, Kairos explained that thermophysical properties and 3-D temperature and flow distributions are not identified as separate phenomena but are captured by the 3-D reflector and core model. The NRC staff determined that the phenomena identified in TR section 3.1.3, "Key Thermal Hydraulics Phenomena," adequately capture the physics important for prediction of core material and reflector temperatures.

2.4.1 Local Thermal Non-Equilibrium (LTNE) Porous Media (PM) Model

The PM modeling approach available within STAR-CCM+ is used to model mass, momentum, and energy transfer in the pebble bed core. TR section 3.5.1.3, "LTNE Porous Media Formulation," describes the 3-D PM model equations for "fluid-phase" (i.e., Flibe) mass, momentum, and energy transfer and "solid-phase" (i.e., fuel and moderator pebbles) energy transfer. Fuel pebbles of different passes (burnup groups) can be represented in this formulation. The formulation assumes a macroscopic volume-average representation of fluid and solid phases with phases assumed to simultaneously occupy the same computational node. Bed porosity and solid-phase volume fractions are used to quantify the solid- and fluid-phase mass and volume and solid-phase surface area in a computational node. [[REDACTED]] The closure models needed for this formulation include empirical correlations for modeling pressure drop, pebble-to-Flibe convective heat transfer, and pebble bed-to-reflector heat transfer. The heat transfer inside the fuel pebble and TRISO particle layers is calculated using a one-dimensional (1-D) conduction model. TR section 4.2, "STAR-CCM+," states that the closure models for pressure drop, packed bed heat transfer, and 1-D conduction for pebble and TRISO particles are implemented in STAR-CCM+ using [[REDACTED]]

The porosity of a randomly packed bed varies in axial and radial directions. The axial variations are due to changes in the core-to-pebble diameter (D/d) ratio. In the radial direction, the porosity is higher near the wall due to wall effects. TR section 3.5.1.3 states that [[REDACTED]] These calculations are discussed in TR section 5.3, "Thermal-Hydraulics," and its subsections. The NRC staff's evaluation of these calculations is provided in SE section 3.3, "Thermal-Hydraulic Model."

The NRC staff determined that the treatment of core pebble bed porosity in the model is acceptable because of the appropriate benchmarking and uncertainty quantification considerations.

Three empirical closing relations selected for the LTNE PM model include the [REDACTED] [REDACTED] The NRC staff notes that the [REDACTED] [REDACTED] was developed for the high-temperature gas-cooled reactor (HTGR). [REDACTED] [REDACTED] TR section 3.5.1.1, "Pressure Drop Correlation," identifies the [REDACTED] [REDACTED] TR section 3.5.1.1 also identifies [REDACTED] [REDACTED]

The [REDACTED] [REDACTED] (see TR table 3-1, "Example of Thermal Hydraulic Parameters of KP-FHR Test Reactors"). [REDACTED] [REDACTED]

The [REDACTED] [REDACTED] This correlation was developed based on experiments performed with air and has been widely used for calculation of pebble bed wall-to-fluid heat transfer. Evaluation of this correlation for prediction of [REDACTED] [REDACTED] is presented in TR section 5.3.

The NRC staff determined that the empirical closing correlations selected for the LTNE PM model have a appropriate pedigree for the application presented in the TR.

TR section 3.5.1.3 describes the modeling of boundary conditions at the interface between the core LTNE PM model and the reflector CFD [REDACTED] [REDACTED] model. At this interface, [REDACTED] [REDACTED] The NRC staff considers this to be a reasonable assumption for normal steady-state conditions because the [REDACTED] [REDACTED] TR section 3.1.3 also states that, if necessary, [REDACTED] [REDACTED] Therefore, the NRC staff determined that the overall treatment of boundary conditions at the LTNE PM model and CFD [REDACTED] [REDACTED] reflector model interface is acceptable.

Fluence and temperature can affect the thermophysical properties and geometries of the core materials (i.e., fuel and moderator pebbles and TRISO particles) and the reflector. Fuel burnup also impacts the thermophysical properties of the fuel. TR section 3.1.3 states that such effects on the thermophysical properties and geometries can be incorporated by [[[REDACTED]

]] The NRC staff determined this methodology is acceptable. However, the input parameters and geometrical specifications, as well as the uncertainty distributions assigned to these values, must be justified for each application of the methodology and should account for the impact of fluence, temperature, and burnup. Accordingly, the NRC staff imposed **Limitation and Condition 2** of this SE.

The NRC staff determined that the TR approach for modeling mass, momentum, and energy transport in the KP-FHR core is acceptable due to:

- the fundamental nature of the porous media model equations;
- prior applications of the porous media model for the packed bed mass, momentum, and energy transfer in literature (References 10 and 11);
- first principle-based 1-D conduction model for heat transfer in core pebbles and TRISO particles that requires only geometry and thermophysical property inputs;
- the pedigree of the closing relations selected for pebble bed pressure drop and heat transfer; and
- comparison of results using this approach against results using the high-fidelity CFD model in TR section 5.3.1 and appendix B.2.2.1 and evaluated in SE section 3.3.1.

2.4.2 Reflector [[[REDACTED]]] Model

The mass, momentum, and energy transport for Flibe in the reflector and the conduction heat transfer in the reflector solid domain (i.e., graphite blocks) are modeled using [[[REDACTED]

]] TR section 3.5.2, "Reflector Modeling," describes the inputs needed for the solid and fluid domain in the reflector model. These include [[[REDACTED]

]] TR section 3.5 states that the use of the [[[REDACTED]]]. The NRC staff determined that this approach is acceptable for modeling Flibe flow in the reflector and heat transfer for the following reasons:

- [[[REDACTED]]];
- steady-state, single-phase liquid flow (no phase change);
- [[[REDACTED]]];

- [[REDACTED]]]; and
- [[REDACTED]].

However, the use of a general-purpose CFD code such as STAR-CCM+ that is not directly validated against applicable CFD-grade data¹ necessitates a more fundamental qualification of the modeling approach. This includes the qualification of approaches used to define problem domain, mesh, boundary conditions, turbulence model, and numerical convergence. Further evaluation of the reflector CFD [[REDACTED]] model problem domain, mesh, boundary conditions, turbulence model selected, and numerical convergence is presented in SE section 2.4.3, “CFD Models for Reflector and Validation of PM Model.”

2.4.3 CFD Models for Reflector and Validation of PM Model

The CFD [[REDACTED]] model is used for two purposes in the TR:

- to model the reflector as a part of the main thermal-hydraulic model, [[REDACTED]]
- [[REDACTED]].

The NRC staff’s evaluation of the CFD [[REDACTED]] model for both applications is presented in this section.

2.4.3.1 Definition of Problem Domain

One of the key benefits of using the CFD modeling approach is the ability to account for the impact of complex geometric features on the flow, heat transfer, and FOMs. These features are physical boundary conditions that guide and interact with the flow. It is common to develop a CFD model domain using detailed drawings or CAD files. However, simplifications are sometimes needed to facilitate practical and efficient computational mesh designs. Since the geometry forms a boundary condition for the CFD model, the impact of any simplifications or assumptions related to the geometry on the FOMs should be quantified.

TR section 3.5.4.1, “Geometric modeling approach and simplifications,” describes the geometrical modeling approach used for the CFD models. The TR states that the need for geometrical simplification is minimized and provides a list of fluid and solid regions represented in a typical integral core and reflector CFD model [[REDACTED]].

The TR also states that the level of geometrical details and simplification will depend on the scope and objective of the problem being analyzed, and sensitivity studies will be performed to evaluate the effects of any geometrical simplifications on the FOMs. The TR further states that, if necessary, the impact of geometric simplifications will be included in the uncertainty estimations for the FOMs.

¹ Applicable CFD-grade data is the experimental data measured in a scaled test facility using the same or applicable test fluid and operating conditions.

The NRC staff found this approach acceptable because it ensures that the geometry in the CFD model is representative of each family of KP-FHR designs (i.e. KP-FHR design with similar geometry, power density and flow conditions) and the impact of any simplifications, if made, are quantified with sensitivity studies that are used to [[REDACTED]]

2.4.3.2 Material Properties

Material properties can impact pressure drops, heat transfer, and buoyancy effects in natural circulation flows. TR section 3.5.4.2, "Material properties," states that [[REDACTED]] The TR further states that the material properties will be consistent with the Kairos Power material qualification program.

The use of [[REDACTED]] However, as discussed in section 2.4.1 and **Limitation and Condition 2** of this SE, the material property inputs for all the core materials and reflector and the geometrical specifications, along with their uncertainty distributions, should be justified and account for the impact of fluence, temperature, and burnup.

2.4.3.3 Mesh

A CFD model's equations are solved on a computational mesh that affects the predicted results. It is important to understand the impacts of the mesh on the calculated FOMs and whether they are significant in light of other uncertainties. Confidence is established in the mesh by demonstrating quality of the mesh design and that the results are adequately converged. This is typically done by following mesh design best practice procedures and completing a grid convergence study in which predictions are made on successively smaller meshes to quantify convergence.

The approach described in TR section 3.5.4.3, "Mesh Design Approach," to develop the mesh design for the CFD models includes:

- [[REDACTED]]
- [[REDACTED]]
- [[REDACTED]]

In addition, the TR states that [[REDACTED]] The NRC staff notes that a mesh design is specific to a particular scenario since mesh refinement is needed in regions of high gradients that can change magnitude and location based on scenario-specific assumptions. The TR notes that [[REDACTED]] The TR provides quantitative metrics for determining acceptability of mesh design.

The NRC staff determined that the mesh design approach described in TR section 3.5.4.3 is consistent with CFD best practice guidance (References 12 and 13) and provides confidence

that an adequate mesh design will be developed for the CFD models for each family of KP-FHR design. Furthermore, the application of the [(REDACTED)] Therefore, this mesh design approach is acceptable for the CFD models used in the TR. However, as indicated in TR section 3.5.4.3, the process for selection and justification of mesh design must be followed for each family of KP-FHR designs and expected operating conditions. Accordingly, the NRC staff imposed **Limitation and Condition 3** that requires justification for the mesh selected for the CFD model using the process described in TR section 3.5.4.3 for each family of KP-FHR designs and expected operating conditions.

2.4.3.4 Boundary Conditions and Source Terms

Boundary conditions, including volumetric source terms, are an integral part of a CFD model used for calculation of mass, momentum, and energy transport. The boundary conditions establish and/or drive the flow and heat transfer mechanisms in the calculation domain. Since the boundary conditions and source terms can significantly impact the calculated FOMs, their uncertainties should be included in the uncertainty evaluation. Furthermore, complete documentation of the boundary conditions is a standard part of model documentation.

TR section 3.5.4.4, "Boundary Conditions and Energy Source Terms," and tables 3-6 through 3-12 describe the boundary conditions and energy source terms for the thermal-hydraulic and CFD models. The tables provide information related to the basis, assumption, or source for the boundary condition. Some of the boundary conditions are selected based on [(REDACTED)]

[(REDACTED)] The NRC staff determined that the approach used in the TR for defining the boundary conditions and the energy source terms is acceptable because it adequately addresses the uncertainties in the source terms and is clearly documented.

2.4.3.5 Turbulence Model

The turbulence modeling approach used in a CFD model is a key attribute for model qualification. In conjunction with the turbulence model selection, an appropriate wall modeling approach is needed to account for the high-gradient region near the wall where shear stress and wall heat transfer are predicted. [(REDACTED)] It is common to benchmark a CFD model with experimental data to support the selection of a turbulence modeling approach.

However, the TR does not provide any direct benchmarking of the CFD models with applicable experimental data. Instead, TR section 3.5.4.5, "Turbulence and Wall Modeling Approach," provides a process that includes assessment of the turbulence model impact on FOMs using sensitivity studies, assessment of [(REDACTED)] and the adoption of best practice guidelines in the wall mesh sizing for the prediction of near-wall flow velocity and thermal gradients.

In addition, the TR states that [(REDACTED)] TR figure 3-10, "Example of turbulence model sensitivity analysis compared with other sources of uncertainty," shows [(REDACTED)]

[REDACTED]

In addition, the TR discusses that the low flow rates, and the associated low turbulence levels in the flow, [REDACTED] Finally, as described in TR section 5.3, [REDACTED]

TR section 3.5.4.5 describes the [REDACTED] and a wall modeling approach where the boundary layer is [REDACTED] The TR notes a first wall cell thickness of [REDACTED] in dimensionless wall units. The NRC staff notes that this turbulence modeling approach and wall mesh sizing is appropriate because of the [REDACTED] number flows associated with the KP-FHR design.

Given the example turbulence model sensitivity and the [REDACTED] in the domain, along with the large margin and plans for startup testing, the NRC staff found the approach presented in TR section 3.5.4.5 acceptable. To ensure adequate implementation of the approach, the NRC staff imposed **Limitation and Condition 4** that requires selection and justification of turbulence and wall models for the CFD models using the process described in TR section 3.5.4.5 for each family of KP-FHR designs and expected operating conditions.

2.4.3.6 Numerical Convergence

Documentation of the numerical solution procedures is important to ensure that solution convergence is adequate for the model. This process involves selecting appropriate differencing schemes and monitoring parameters to ensure accuracy and convergence of the solution.

TR section 3.5.4.6, "Numerical Solution," notes that second-order numerical schemes are used for the CFD [REDACTED] model, and numerical solution verification includes iterative convergence assessments. These assessments include monitoring the global mass and energy balance as well as the vessel outlet temperature, pressure drops, Nusselt numbers, and mass flow rates at selected interfaces. In addition, the TR notes local monitoring of velocity and temperatures at selected points as well as the wall y^+ values.

The NRC staff notes that this approach follows the CFD best practice guidelines for differencing schemes and solution monitoring. The NRC staff determined that the approach, [REDACTED] provides confidence that the solution gradients are adequately resolved and that solutions are adequately converged.

2.5 Overall Methodology Process

The role of key wrapper tools (KACEGEN, KPACS, and KPATH) in the overall methodology calculation framework is evaluated in this section. KACEGEN is Kairos's CE nuclear data library generation code, which calls Los Alamos National Laboratory's NJOY21 software to generate ACE-formatted libraries for Serpent 2 and MCNP at a range of temperatures to account for changes in thermal scattering and resonance absorption with temperature. The NRC staff reviewed the temperature range, temperature intervals, and specific evaluated nuclear data library versions (e.g., ENDF/B-VII.1, ENDF/B-VIII.0, JEFF 3.3) evaluated by the applicant and

determined that the nuclear data approach described in the TR is adequate because the data cover the range of expected KP-FHR conditions.

KPACS provides predictions of the operational history of the reactor core including the effects of fuel and moderator pebble loading, re-circulation, and discharge. It can be used [(

)] As identified in TR section 7.2 Limitation 7, the pebble velocity needs to be a small fraction of the time constant of delayed neutron precursors for this approach to be applicable. The geometry is passed to Serpent 2 to evaluate neutronic FOMs and burnup, with an explicit geometrical representation of the pebbles and TRISO particles within the bed. The particle isotopics are tracked within groups of pebbles residing in the same spectral zone and experiencing the same pass through the core. The NRC staff determined that this approach to representing the core is reasonable with an appropriate level of approximation, as it is consistent with both legacy and emerging core physics modeling paradigms within the pebble bed reactor industry, albeit at a higher level of spatial and spectral resolution due to explicit modeling of individual pebbles in the geometric model and the use of continuous energy cross sections. Any major deficiencies in the KPACS approach would be determined during the operation of the test reactor via burnup measurements of pebbles as they are extracted and would accrue gradually with exposure, precluding sudden-onset gross failures that could challenge safety limits.

KPATH provides for coupled steady-state neutronic and thermal-hydraulic core simulation by interfacing Serpent 2 with STAR-CCM+. [(

)] The NRC staff determined that this approach is reasonable because it assures that the relevant physics and fields are accounted for in the model used to determine power and temperature distributions.

2.6 Quality Assurance

TR section 4.6, "Software Quality," states that software and computer codes used in the TR methodology are maintained under the Kairos Power software quality assurance program (TR section 7.2, Limitation 8). The NRC staff notes that the approval of methodologies using computer codes is typically based on specific code versions, theory manuals, and user manuals.

During audit discussions Kairos clarified that STAR-CCM+ version 2021.3 was used for the analyses described in this TR. TR section 4.3, "Serpent 2," provides the version of Serpent 2 used for the calculations provided within the TR and states that the methodology may be applied using future versions of Serpent 2. TR section 3.2, "General Modeling Approach," also states that the transfer of information used as input for applications of this methodology is managed for compliance with appropriate quality assurance requirements. TR section 4.3 further describes the process for managing and evaluating applicability of future code versions via the Kairos Power software quality assurance program. The adequacy of the software quality assurance program is outside the scope of this TR.

3. Methodology Validation, Verification, and Uncertainty Analysis

3.1 Discrete Element Method

TR section 5.1, "Discrete Element Method (DEM)," states that the pebble velocity profiles calculated by DEM [[REDACTED]]. As discussed in SE section 2.2, [[REDACTED]]. The buoyancy force is mainly dependent on pebble geometry and pebble and coolant densities. Therefore, the primary sources of uncertainty in the DEM model are [[REDACTED]]. The [[REDACTED]] in DEM is calculated from [[REDACTED]]. The validation of [[REDACTED]] is discussed in TR section 5.3 and is evaluated in SE section 3.3.

TR section 5.1 focuses on the process for [[REDACTED]]. The TR justifies this process as sufficient because [[REDACTED]].

TR section B.1, "Discrete Elements Modeling," demonstrates the above [[REDACTED]].

Based on this successful demonstration of the approach and the sensitivity of DEM predictions on core physics parameters, the NRC staff determined that [[REDACTED]] is acceptable for the DEM model.

3.2 Neutronics

The validation, verification, and uncertainty analysis for neutronic FOMs in TR section 5.2 follows the strategy established at the beginning of section 5. TR sections 5.2.1 and 5.2.2 describe code-to-code benchmarking, demonstrating Serpent 2's capabilities in comparison with two independent, well-established MC-based neutronic codes in [[REDACTED]] and [[REDACTED]] to gain confidence in Serpent 2 predictions. To enable consistent comparisons, [[REDACTED]].

The NRC staff determined that the consideration of sources of uncertainties and methodology to develop QU and bias as described in TR sections 5.2.3 and 5.2.4, respectively, are acceptable because they provide reasonably informed estimates and when directly comparable data is available (e.g., data from the Hermes and Hermes 2 test reactors) NRFs may be updated, as described in TR section 5.4 (TR section 7.2, Limitation 3). The NRC staff also has reasonable assurance that the DCs described in TR section 5.2.5 and table 5-27, "Sources of Unquantified Uncertainty," are acceptable for the Hermes and Hermes 2 test reactors because they provide sufficient additional margin such that core design-related safety limits will be ensured during operation (TR section 7.2, Limitation 5).

As stated in TR section 7, "Summary," the predictions of the neutronic module will be confirmed during the fuel loading process and subsequent zero-power testing of Hermes and future KP-FHR reactors. TR section 6.6, "Startup Physics Testing," describes how the startup process of a KP-FHR will use nuclear design calculations. Critical mass predictions from Serpent 2 will be used during fuel loading. After achieving criticality, zero-power testing, including control rod calibration, shutdown rod worth, and isothermal temperature coefficient measurements, will be performed and compared against predictions. If the measured values lie outside of the predicted values with uncertainty bands, testing will be suspended and impacts on the safety analysis will be evaluated before any further testing is performed. Measurements will also be taken and compared to predictions during power ascension. The NRC staff notes that this is a reasonable approach to startup physics testing and is consistent with standard industry practice. The NRC staff will review specific testing plans and procedures as part of future licensing submissions.

The strategy for updating the NRFs based on operational data described in TR section 5.4 uses standard confidence interval analysis and is therefore acceptable to the NRC staff. The NRF update process includes the use of test data to formally quantify uncertainties for a system that is directly applicable to future KP-FHRs (TR section 7.2, Limitation 3). The NRC staff expects to review these data prior to approving application of this methodology for future KP-FHR power reactors. Therefore, the NRC staff imposes **Limitation and Condition 5**, requiring the operating data gained as part of validating the methodology described in this TR to be reviewed by the NRC staff and any updates to NRFs to be reviewed and approved by the NRC staff prior to future applications to power reactor systems.

TR section 6 describes calculations performed in nuclear design that support subsequent analyses (e.g., safety analysis, source term analysis). These include such quantities as reactivity coefficients, integral and differential control and safety rod worths, kinetics parameters, power distribution, stability, material depletion and transmutation, and radiation fluence. Although the NRC staff does not make any determinations on the information in this section, the NRC staff generally found that the list of evaluated quantities and approaches to calculate primary and derived parameters is reasonable.

TR appendix A provides example implementation of the core design methodology for a version of the Hermes design, including detailed demonstrations of the calculations described in TR section 6 and the uncertainty and bias estimation described in section 5. Because this appendix is presented primarily to illustrate the methods and assist the staff in understanding how the methodology described in the TR is implemented, the staff did not make any determinations on TR appendix A.

3.3 Thermal-Hydraulic Model

3.3.1 Thermal-Hydraulic Model Validation

TR section 5.3, "Thermal-Hydraulics," describes validation, verification, and uncertainty quantification of the thermal-hydraulic model. Validation of the thermal-hydraulic model is primarily focused on the [REDACTED]

As discussed in SE section 2.4.1, "Local Thermal Non-Equilibrium (LTNE) Porous Media (PM) Model," the pebble bed pressure drop, and heat transfer empirical correlations implemented in the LTNE PM model were developed based on experimental data that is representative of high-temperature gas-cooled reactors. The typical approach to extend the applicability of empirical correlations to new design and operating conditions would be to validate the correlations against applicable separate effects test (SET) data. In the absence of data representative of KP-FHR designs and conditions, these correlations are assessed [REDACTED]

[REDACTED] TR section 5.3 states that this validation approach is commensurate with the applications of the thermal-hydraulic model in the methodology. The TR describes that [REDACTED]

The numerical validation includes use of [REDACTED]

[REDACTED] Figure 5-13, "Geometrical Features of the High-Fidelity Packed Bed Model," of the TR shows an example of [REDACTED]

[REDACTED] as described in TR section 5.3.2, "Uncertainty Quantification."

TR section 5.3.8, "Applicability of Porous Media Correlations," describes the [REDACTED]

[REDACTED] An example implementation of this process for [REDACTED] is presented in TR sections B.2.1.1, "Pressure Drop" (and table B.2-1, [REDACTED]) and B.2.1.2, "Pebble-Flibe Heat Transfer Coefficient" (and table B.2-2, [REDACTED])

TR section B.2.2.1, [REDACTED] and its associated tables, provides comparisons of [REDACTED]

Similar results for [REDACTED]

TR section 5.3.8 states that [REDACTED]

[REDACTED] Limitation 6 in TR section 7.2, "Limitations," refers to the methodology in TR

section 5.3.8 for determining ranges of applicability. The NRC staff includes this limitation through **Limitation and Condition 1** of this SE.

Sensitivity of Methodology FOMs to Thermal-Hydraulic Model Output Parameters

TR section 7, "Summary," notes that [[[REDACTED]]]. This statement is supported by the results of example sensitivity calculations discussed in TR sections A.1.1.8, "Power Peaking Factors," and A.1.2.2, "Demonstration of Sensitivity Study." As discussed in section A.1.1.8, [[[REDACTED]]]. TR Figure A.1-17, [[[REDACTED]]]. Similarly, figures A.1-13, "Sensitivity of Effective Multiplication Factor with Respect to Pebble/TRISO Boron, Reflector Density, and Flibe Density and Temperature," and A.1-18, "Sensitivity of Effective Multiplication Factor with Respect to Flibe, TRISO-Kernel, and Reflector Temperatures," of the TR show [[[REDACTED]]].

Large Safety Margin for the Methodology FOMs

Table 6-5, "Steady state Hermes predictions compared to selected reference FOM limits," of the TR shows an example of the methodology FOMs calculated for the Hermes test reactor compared to the applicable limits. The table shows that there is significant margin for the important safety parameters such as fuel and reflector temperatures and shutdown margin.

Use of Startup Test Data and Confirmatory SET Data for Validation

TR section 5.3.7, "KP-FHR Testing," confirms that, for the Hermes test reactor, there will be monitoring and measurement of reflector graphite, core outlet, and reactor vessel inlet and outlet temperatures. The NRC staff notes that this will provide validation of the methodology to confirm that the temperatures are predicted within the calculated uncertainty bound. In addition, the direct monitoring of the temperatures facilitates operational safety.

As discussed in SE Section 3.2, "Neutronics," [[[REDACTED]]]. Furthermore, TR section 6.7, "Example Limits and Margins for Selected Applications," states that [[[REDACTED]]]. These measurements, some of which may be used to update the NRFs, provide additional validation of the methodology.

TR section 5.3 clarifies that [[[REDACTED]]]. The NRC staff determined this to be acceptable provided the methodology incorporates conservatism sufficient to bound the test data.

Qualification of the CFD [[[REDACTED]]]. Model Based on CFD Best Practices

TR section 3.5.4, “[REDACTED] Computational Model Design and Implementation Example,” describes the qualification of the CFD [REDACTED] model. The NRC staff’s evaluation of the CFD [REDACTED] model qualification process and limitations on its use are discussed in SE sections 2.4.2 and 2.4.3. The use of CFD best practices to define problem domain, mesh, boundary conditions, turbulence model, and numerical convergence allow development of a well-qualified CFD model for numerical validation.

In summary, the NRC staff determined that the numerical validation approach proposed in the TR is adequate and acceptable for the validation of the thermal-hydraulic model due to the:

- limited sensitivity of the methodology FOMs to the thermal-hydraulic model-calculated parameters;
- large safety margins for the methodology FOMs;
- use of startup and operational test data for limited validation of the methodology;
- use of confirmatory validation against the Kairos KP-FHR SET data; and
- use of a well-qualified CFD-RANS model based on the CFD best practices.

3.3.2 Thermal-Hydraulic Model Uncertainty Quantification

TR section 5.3.2, “Uncertainties Quantification,” describes the methodology for quantification of uncertainties in thermal-hydraulic model FOMs (i.e., core material and reflector temperatures). TR equation 43 and figure 5-14, “Porous Media Validation Framework,” summarize the uncertainty quantification approach. As shown in equation 43, the methodology best-estimate value for a FOM is [REDACTED]

[REDACTED] Limitation 4 in TR section 7.2 states that these bias and confidence level factors are limited to the KP-FHR test reactors (i.e., Hermes and Hermes 2), and their use in power reactors will be justified based on applicable SET data or measured data from the test reactors. The NRC staff includes this limitation through **Limitation and Condition 1** of this SE.

The high-fidelity CFD [REDACTED] model results are [REDACTED]

The model input uncertainty is calculated from the propagation of uncertainties in different model input parameters. As identified in TR section 5.3.5, “Input Uncertainties,” the inputs considered for the uncertainty propagation include [REDACTED] The numerical uncertainty described in TR section 5.3.4, “Numerical Error and Solution Verification,” is calculated using the CFD best practices for the estimation of numerical errors and solution verification.

TR section A.2, “Thermal Hydraulics,” provides example values for estimated uncertainties for the thermal-hydraulic model FOMs. TR table A.2-1 and A.2-2 provide example biases and uncertainties for the Flibe, kernel, and reflector temperatures, while figure A.2-1 shows input parameter uncertainties for the Flibe and TRISO kernel temperatures.

The NRC staff determined that the uncertainty quantification methodology is acceptable because it adequately accounts for the major sources of uncertainty. Furthermore, the proposed values for the bias factor and confidence interval factors provide a conservative estimation of the uncertainty band. Although the methodology proposed for the propagation of input uncertainties is reasonable, as discussed earlier in SE section 2.4.1, the methodology input parameters and geometrical specifications and the uncertainty distributions assigned to these parameters must be justified for each application of the methodology and should account for the impact of fluence, temperature, and burnup, as reflected in **Limitation and Condition 2**.

LIMITATIONS AND CONDITIONS

The NRC staff applies the following additional limitations and conditions on the acceptance of this TR:

1. Any licensing application referencing this TR must demonstrate that the limitations listed in TR section 7.2 are met, subject to NRC staff review and approval.
2. Any licensing application referencing this TR must provide for NRC staff review and approval acceptable justification that the input values for thermophysical properties and geometrical specifications, as well as the uncertainty distributions assigned to these values, for the core materials and reflector adequately account for the impact of fluence, temperature, and burnup.
3. Any licensing application referencing this TR must provide for NRC staff review and approval acceptable justification that the mesh design for the CFD models is performed using the process described in TR section 3.5.4.3 for each family of KP-FHR designs and expected operating conditions.
4. Any licensing application referencing this TR must provide for NRC staff review and approval acceptable justification that the selection of turbulence and wall models for the CFD models is performed using the process described in TR section 3.5.4.5 for each family of KP-FHR designs and expected operating conditions.
5. Any licensing application for a KP-FHR power reactor referencing this TR must provide for NRC staff review supporting test data obtained from operation of the Hermes test reactors or other representative data used for final validation and justification of its applicability for each family of KP-FHR designs and expected operating conditions. In addition, if updates are made to nuclear reliability factors per the TR methodology, the changes will be provided for NRC staff review and approval.

CONCLUSION

The NRC staff determined that Kairos's KP-TR-024-P, "KP-FHR Core Design and Analysis Methodology," Revision 1 provides an acceptable methodology for the calculation of parameters in TR section 3.6 and for the quantification of biases and uncertainties subject to the limitations and conditions discussed above. The evaluation of final compliance or conformance with the identified regulations and PDCs will be performed during the review of a licensing application referencing this TR.

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Section B



KP-TR-024-NP-A

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KP-FHR Core Design and Analysis Methodology

Topical Report

Revision No. 1
Document Date: June 2025

Non-Proprietary

KP-FHR Core Design and Analysis Methodology			
Non-Proprietary	Doc Number	Rev	Effective Date
	KP-TR-024-NP-A	1	June 2025

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KP-FHR Core Design and Analysis Methodology			
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Rev	Description of Change	Date
0	Initial Issuance	April 2024
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KP-FHR Core Design and Analysis Methodology			
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Executive Summary

Kairos Power is pursuing the design, licensing, and deployment of a Fluoride Salt-Cooled, High Temperature (KP-FHR) reactor. To enable these objectives, the development of a technology-specific core design and analysis methodology is required. This report describes the methodology for core physics, thermal hydraulic analysis, and radiation effects on materials of the KP-FHR.

The core design methodology described within this report applies to the steady-state operation of a KP-FHR. This methodology is informed by research publications pertaining to FHRs and other pebble bed reactors. The methods are also informed by key neutronics and thermal hydraulic steady state phenomena in the KP-FHR.

The KP-FHR core design methodology is composed of the Serpent 2 nuclear design and STAR-CCM+ thermal, fluid, and discrete element modeling design codes. These codes are connected by a series of Kairos Power developed wrapper codes. The verification and validation (V&V) methodology for Serpent 2 and STAR-CCM+ codes is also described herein.

Serpent 2 and STAR-CCM+ and the associated wrapper codes are used to calculate core composition during various phases of operation and to calculate corresponding parameters such as core reactivity coefficients, control and shutdown element worth, shutdown margin, power distribution, radiation damage within and including reactor vessel, core and reflector temperature distributions. The methodology for using the codes to perform these calculations and the limitations on the use of this methodology are provided.

Additionally, this topical report presents a structured approach for quantifying uncertainty, using known uncertainties in input parameters, and capturing biases pertinent to nuclear data. A discussion is provided on informed biases and additional discretionary conservatism (DC) for defining nuclear reliability factors (NRFs).

Sample problems are provided to illustrate the methodology for performing core design calculations and determining uncertainties.

KP-FHR Core Design and Analysis Methodology			
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NOMENCLATURE

List of Abbreviations	
ACE	a compact ENDF
AHTR	advanced high-temperature reactor
ARI	all-rods in
ARO	all-rods out
CE	continuous energy
CFD	computational fluid dynamics
CHM	carbon to heavy metal atom ratio
COF	coefficient of friction
CRAM	Chebyshev Rational Approximation Method
CSAS	criticality safety analysis sequence
CTC	coolant temperature coefficient
CVC	coolant void coefficient
DC	discretionary conservatism
DEM	discrete elements method
DHRS	decay heat removal system
EBC	equivalent boron concentration
EMDAP	evaluation model development and assessment process
ENDF	evaluated nuclear data files
FHR	fluoride-cooled high temperature reactor
FOM	figure-of-merit
FTC	fuel temperature coefficient
[[]]	[[]]
HTGR	high temperature gas-cooled reactor
IET	integral effect test
IPyC	inner pyrolytic carbon layer
ITC	isothermal temperature coefficient

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List of Abbreviations	
KERMA	kinetic energy released per unit mass
KP-FHR	Kairos Power Fluoride Salt-Cooled, High Temperature Reactor
KPACS	Kairos Power Advanced Core Simulator
KPATH	Kairos Power Advanced Thermal Hydraulics
LTNE	local thermal non-equilibrium
MG	multi-group
MTC	moderator temperature coefficient
NRC	Nuclear Regulatory Commission
NRF	nuclear reliability factor
OPyC	outer pyrolytic carbon layer
PDC	principal design criteria
PEM	pebble extraction machine
PIL	pebble insertion line
PIRT	Phenomena Identification and Ranking Table
PSP	primary salt pump
QA	quality assurance
RCS	reactivity control system
RSS	reactivity shutdown system
RTC	reflector temperature coefficient
SARRDL	specified acceptable system radiological release limit
SET	separate effect test
SiC	silicon carbide
sKPH	simplified KP-FHR model
SST	shear stress transport
TH	thermal hydraulics
[[]]	[[]]
TRISO	tri-structural isotropic
TSL	thermal scattering law

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List of Abbreviations	
UA	uncertainty analysis
UQ	uncertainty quantification
V&V	verification and validation

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1 INTRODUCTION

Kairos Power is pursuing the design, licensing, and deployment of the Kairos Power Fluoride Salt-Cooled, High Temperature Reactor (KP-FHR) technology, including non-power test reactor and commercial power reactor designs. To support these objectives, Kairos Power has developed a core design and analysis methodology applicable to the KP-FHR design.

This topical report describes the methodology for core design and analysis of the KP-FHR design during startup, power ascension, and equilibrium conditions. The report describes the methodology used to model and analyze the nuclear and thermal-hydraulic behavior of the KP-FHR core during steady-state conditions. The core design and analysis methodology is used to calculate reactivity coefficients, rod worth, power distribution, temperature distribution, flux distribution, kinetics parameters, material depletion, radiation damage and heating. These output parameters are used for other applications including fuel performance, source term, and safety analysis. The application of these methods is illustrated in an example calculation provided for demonstration purposes.

The primary tools used to model the core include the Serpent 2 and STAR-CCM+ codes and the calculational methods associated with them, which are described in this topical report. V&V of the Serpent 2 and STAR-CCM+ codes is performed through higher fidelity methods and code-to-code benchmarks. This topical report also includes a methodology to quantify the impacts of quantifiable uncertainties and to determine bias and discretionary conservatism to inform nuclear reliability factors (NRFs). This method ensures that calculated values are reported with sufficiently conservative uncertainty values to be used as input to safety analysis for the KP-FHR. The final confirmation of criticality and other core design parameters is performed during the fuel loading process and zero power testing.

Kairos Power requests Nuclear Regulatory Commission (NRC) review and approval of this topical report for the following:

- The use of Serpent 2, STAR-CCM+ and the calculational framework to calculate the parameters and figures of merit summarized in Section 3.6.
- The calculational methodology used to determine quantifiable uncertainties described in Section 5.2.3, the biases described in Section 5.2.4 and Table 5-26, and the discretionary conservatism described in Section 5.2.5 and Table 5-27.
- The validation methodology for porous media modeling and the applicability of closure models through higher fidelity computational fluid dynamics (CFD) models described in Section 5.3 and the bias and confidence intervals stated in Section 5.3.6 to bound temperature calculations for over/underestimation.
- The methodology presented in Section 5.4 to update nuclear reliability factors (NRFs) using operational data.

Further regulatory assessment applicable to this core design and analysis methodology is provided in this section of this topical report. Section 2 of the report provides general design features of a typical KP-FHR. These features are modeled through three domains: the motion of pebbles in the core (discrete element modeling (DEM)), neutronics of the core and thermal hydraulics. These modeling paradigms are described in Section 3 of this topical report. Section 4 of the report provides a summary of the codes used for modeling the KP-FHR. These codes are qualified for use in accordance with the methodology

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described in Section 5. Section 6 provides the application of this methodology and how figures of merit and certain outputs are used for safety analysis, nuclear design and thermal hydraulic analysis. Appendices A and B with example calculations are provided to illustrate the use of this methodology.

1.1 Design Features

1.1.1 Design Background

To facilitate NRC review and approval of this report, design features considered essential to the KP-FHR technology are provided in this section. These key features are not expected to change during the ongoing detailed design work by Kairos Power and provide the basis to support the safety review. Should fundamental changes occur to these design features or revised regulations be promulgated that affect the conclusions in this report, such changes will be reconciled and addressed in future license application submittals.

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

1.1.2 Key Design Features of the KP-FHR

The KP-FHR is a high temperature reactor with molten fluoride salt coolant operating at near-atmospheric pressure. The fuel in the KP-FHR is based on the tri-structural isotropic (TRISO) high-temperature, carbonaceous-matrix coated particle fuel which was originally developed for high temperature gas-cooled reactors (HTGRs) in a pebble fuel element. Coatings on the particle fuel provide retention of fission products. The reactor coolant is a chemically stable molten fluoride salt mixture, 2 LiF: BeF₂ (Flibe), which also provides retention of fission products. A primary coolant loop circulates the reactor coolant using pumps and transfers the heat via a heat exchanger. The design includes decay heat removal for both normal conditions and postulated event conditions. Passive decay heat removal, along with natural circulation in the reactor vessel, is used to remove decay heat in response to a postulated event. The KP-FHR does not rely on electrical power to achieve and maintain safe shutdown for postulated events.

Instead of the typical light water reactor (LWR) low-leakage, pressure retaining containment structure, the KP-FHR design relies on a functional containment approach similar to the Modular High Temperature Gas-Cooled Reactor (MHTGR). The KP-FHR functional containment safety design objective is to meet 10 CFR 50.34 (10 CFR 52.79) offsite dose requirements at the plant's exclusion area boundary with margin. A functional containment is defined in Regulatory Guide (RG) 1.232, "Guidance for Developing Principal Design Criteria for Non-Light water Reactors" as a "barrier, or set of barriers taken together, that effectively limit the physical transport and release of radionuclides to the environment across a full range of normal operating conditions, anticipated operational occurrences, and accident conditions." As also stated in RG 1.232, the NRC has reviewed the functional containment concept and found it "generally acceptable," provided that "appropriate performance requirements and criteria" are developed. The NRC staff has developed a proposed methodology for establishing functional containment performance criteria for non-LWRs, which is presented in SECY-18-0096, "Functional

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Containment Performance Criteria for Non-Light-Water-Reactors.” This SECY document has been approved by the Commission.

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

1.2 Regulatory Assessment

The following section provides a brief review of regulatory requirements applicable to reactor core design and analysis for the KP-FHR.

1.2.1 10 CFR Requirements

Nuclear Regulatory Commission (NRC) regulations in 10 CFR 50.34(a)(4), 10 CFR 50.34(b)(4) and 10 CFR 52.79(a)(5) require an analysis and evaluation of the design and performance of structures, systems, and components of the facility. These regulations assess the risk to public health and safety resulting from operation of the facility and determine the margins of safety during normal operations and transient conditions during the life of the facility, and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents.

Consistent with these requirements, the KP-FHR core design and analysis methodology described in Sections 2.2, 2.3, 3, 4, and 5 is used to analyze the fuel and core during normal operation to determine margins of safe operation of the reactor core. Computational outputs of the methods as listed in Section 3.6 and described further in Section 6, are used to further determine the adequacy of structures, systems and components that provide for the prevention and mitigation of postulated events.

1.2.2 Principal Design Criteria for the Reactor Core

The principal design criteria that apply to KP-FHR reactors are contained in the “Principal Design Criteria for the Kairos Power Fluoride Salt-Cooled High Temperature Reactor Topical Report” (Reference 1). The core design and analysis methods are used to determine the margins of safe operation for the reactor core during normal operation to demonstrate compliance with the following principal design criteria (PDCs):

PDC 10, Reactor design

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

Together, the modeling paradigm described in Section 3.2, the modeling tools STAR-CCM+ and Serpent 2 described in Section 4, and the validation, verification and uncertainty analyses described in Section 5 provide a methodology to support designing and modeling the reactor core and the establishment of normal operational safety margins. These safety margins established for normal operation are the basis for safety analyses used to develop and determine margins to safety during postulated events described

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in safety analysis reports. The methods in Section 3 and the modeling tools in Section 4 of this topical report are used to establish safety limits for normal operation by determining the reactivity feedback, shutdown margin, rod worth, power distribution, temperature distribution, flux distribution, kinetics parameters, material depletion, and helium generation in the KP-FHR core. Development of this methodology to demonstrates that reactor core safety margins are not exceeded, which satisfies, in part, PDC 10.

PDC 11, Reactor inherent protection

The reactor core and associated systems that contribute to reactivity feedback shall be designed so that, in the power operating range, the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

The modeling paradigm described in Section 3.2, the modeling tools described in Section 4, and the validation and benchmarking analyses described in Section 5.2 provide a means of demonstrating that the net effect of prompt inherent nuclear feedback on the reactor core design compensates for rapid increases in reactivity in the KP-FHR core. Methods used to determine the reactivity feedback and kinetics parameters are described in Section 4 and the qualification of those methods is described in Section 5 via benchmarking studies. Conservative uncertainties and biases are added to calculational results, thereby providing reasonable assurance that the methodology can provide results for demonstrating compliance with PDC 11.

PDC 12, Suppression of reactor power oscillations

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

The methodology described in the topical report supports PDC-12 determination by calculating the neutronic properties (diffusion length and migration area) and reactivity coefficients, which contribute to preventing or dampening oscillations. Nuclear instability initiated by reactor control is not addressed by the methodology in this topical report. The initiators of power oscillations are mitigated by strong neutronic coupling in the core (long neutron diffusion length), negative reactivity feedback, and single-phase coolant. Final PDC 12 compliance will be demonstrated in an operating license application.

1.2.3 Principal Design Criteria for Other Structures, Systems and Components

Several calculational outputs of the methods provided in this topical report are used in safety analysis reports to further demonstrate the adequacy of structures, systems, and components that prevent and mitigate postulated events. The core design and analysis methodology described herein provides margins to safety during normal operation and these safety margins are used as inputs (i.e., initial conditions) for safety analyses that determine margins to safety during postulated events.

PDC 16, Containment design

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

The KP-FHR uses TRISO fuel particles (described in Section 2.1.1) and Flibe coolant (described in Section 2.1.2), the combination of which makes up the functional containment of the design. The methodology

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described herein is used to design the core such that core design parameters (i.e., reactivity coefficients, shutdown margin, rod worth, kinetics parameters, material depletion, flux distribution, temperature distribution, power distribution, radiation heating and damage) have margins to safety and adequately account for core changes during normal operation due to variations in fuel and coolant. The pebble peaking factor, as an output from the core design and analysis methodology, is also used to assess fuel performance. The core design and analysis methodology therefore can be used, in part, to support a demonstration of compliance with PDC 16.

PDC 25, Protection system requirements for reactivity control malfunctions

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

The core design and analysis methodology is used to provide calculational output (i.e., rod worth and shutdown margin) for core and fuel limits, which supports defining operational safety limits that account for a single malfunction of the reactivity shutdown system. This methodology is used in safety analysis reports which demonstrate in part, compliance with PDC 25.

PDC 26, Reactivity control systems

A minimum of two reactivity control systems or means shall provide:

- (1) A means of inserting negative reactivity at a sufficient rate and amount to assure, with appropriate margin for malfunctions, that the specified acceptable system radionuclide release design limits are not exceeded and safe shutdown is achieved and maintained during normal operation, including anticipated operational occurrences.*
- (2) A means which is independent and diverse from the other(s), shall be capable of controlling the rate of reactivity changes resulting from planned, normal power changes to assure that the specified acceptable system radionuclide release design limits are not exceeded.*
- (3) A means of inserting negative reactivity at a sufficient rate and amount to assure, with appropriate margin for malfunctions, that the capability to cool the core is maintained and a means of shutting down the reactor and maintaining, at a minimum, a safe shutdown condition following a postulated accident.*
- (4) A means for holding the reactor shutdown under conditions which allow for interventions such as fuel loading, inspection and repair shall be provided.*

The core design and analysis methodology described herein is used to perform analyses that demonstrate that the KP-FHR core has sufficient negative reactivity and shutdown margin to demonstrate in part, compliance with PDC 26.

PDC 28, Reactivity limits

The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to ensure that the effects of postulated reactivity accidents can neither:

- (1) result in damage to the safety significant elements of the reactor coolant boundary greater than limited local yielding nor*
- (2) sufficiently disturb the core, its support structures, or other reactor vessel internals to impair significantly the capability to cool the core.*

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The core design and analysis methodology is used to provide core output parameters to determine reactivity control limits. Reactivity control limits protect the fuel and other reactor systems from a hypothetical rapid change in reactivity and power. Rapid changes in power and large temperature increases could cause damage to the reactor coolant flow path and/or reactor coolant boundary and impair core cooling. The core design and analysis methodology provides a means of demonstrating the margins to operating safely during normal conditions to preclude the adverse impacts of a hypothetical rapid reactivity increase. The methodology therefore can be used to demonstrate compliance with PDC 28.

PDC 31, Fracture prevention of reactor coolant boundary

The safety significant elements of the reactor coolant boundary shall be designed with sufficient margin to ensure that when stressed under operating, maintenance, testing, and postulated accident conditions, (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures, service degradation of material properties, creep, fatigue, stress rupture, and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation and coolant composition, including contaminants and reaction products, on material properties, (3) residual, steady-state, and transient stresses, and (4) size of flaws.

The core design and analysis methodology is used to demonstrate the radiation damage to stainless steel material (i.e., reactor vessel 316H and weld material) does not cause failure to the reactor coolant boundary. Displacement per atom (DPA) and helium generation calculations support this demonstration and are performed using the calculational method shown in Section 6.2.3. Therefore, the methodology can be used to evaluate the expected effects of irradiation on the reactor coolant boundary and demonstrate in part, compliance with the requirements of PDC 31.

PDC 34, Residual heat removal

A system to remove residual heat shall be provided. For normal operations and postulated events, the system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable system radionuclide release design limits and the design conditions of safety related elements of the reactor coolant boundary are not exceeded.

Suitable redundancy in components and features and suitable interconnections, leak detection and isolation capabilities shall be provided to ensure that the system safety function can be accomplished, assuming a single failure.

The core design and analysis methodology determines the heat transfer for pebble-to-pebble and pebble-to-reflector interactions during normal operation. A predictive model can be used to determine material temperatures in the core thereby demonstrating, in part, compliance with PDC 34.

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2 KP-FHR CORE DESIGN FEATURES

2.1 Reactor Core General Features

The KP-FHR core is a randomly packed pebble-bed with molten fluoride salt coolant operating at high temperature and near-atmospheric pressure. The pebble-bed core fills the space created by stacked graphite blocks, and the graphite blocks maintain a coolable geometry and act as a neutron reflector. The graphite reflector blocks and pebbles are buoyant in the molten fluoride salt coolant. Engineered channels in the reflector are used to direct coolant flow, house instrumentation, and insert control elements. The vessel encloses the core barrel and reflector blocks while maintaining the coolant flow path. Pebbles are circulated through the core to manage the core composition and control excess reactivity.

In a KP-FHR reactor core, there are geometrical design features present that are explicitly captured in the DEM and neutronic analysis, described further in Section 3.3 and 3.4 respectively. These features are shown and labeled in Figure 2-1 and are representative of the typical KP-FHR design. Figure 2-2 shows the explicit geometry modeling in Serpent 2 based on the design features of the typical KP-FHR design shown in Figure 2-1.

The highest fission power density is in the pebble bed core, which is defined as the volume from the top of the converging region to the bottom of the diverging region. Core geometrical characteristics such as the conic regions and the defueling chute are designed to balance neutronic performance, heat removal, and the radial pebble residence time profile. The function of the fueling region, located at the bottom of the reactor core, is to guide the pebbles coming from the insertion line into the reactor core. The fueling region also provides space for the pebbles to enter when displaced by the insertion of the shutdown elements. The defueling chute, located at the top of the reactor core, guides the pebbles to the extraction mechanism, and serves as a low power-density region for the short-lived fission products to decay. This in-vessel decay time reduces the decay heat for the pebble handling and storage system (PHSS) thermal management and reduces the activity of the pebbles for the count-rate requirements of the burnup measurement system.

Engineered channels in the reflector are used to direct coolant flow and house instrumentation. Engineered coolant inlet and outlet channels in the reflector blocks both above and below the core are designed to reduce pressure losses while still achieving acceptable flow distribution and flow rates through the core. The reflector block design is characterized by radial and axial gaps between blocks and at the interface with the core barrel. This geometry causes a portion of the coolant flow to bypass the core region. Additional engineered channels in the reflector are also used to reduce the temperature in the reflector.

The reactivity control system (RCS) consists of control elements that insert into engineered channels in the reflector. The reactivity shutdown system (RSS) consists of shutdown elements that directly insert into the pebble bed. The control elements are used for planned power maneuvers of the KP-FHR reactor. Only the control elements are needed to achieve short-term shutdown (i.e., not considering delayed effects from xenon). To achieve the required safe shutdown conditions, the shutdown elements are inserted (assuming the highest worth shutdown element fully withdrawn).

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2.1.1 Fuel and Moderator Pebbles

The KP-FHR uses the Tri-Structural Isotropic (TRISO) carbonaceous-matrix coated fuel particle design embedded in a pebble form (Figure 2-3). The fuel kernel and the coatings on the particle fuel provide retention of fission products. TRISO particles are dispersed within the fuel pebble’s fuel annulus. The fuel particles can have a range of different enrichments, from depleted uranium up to the upper limit of high assay low-enriched uranium (HALEU) (i.e., 20 wt% U235).

The KP-FHR fuel pebbles are buoyant in reactor coolant under both steady state and postulated events. There are also graphite moderator pebbles in the core to enhance the moderation (i.e., increase the carbon to heavy metal atom ratio [CHM]). In steady-state operations, the pebbles circulate through the core region slowly, and fresh fuel pebbles are inserted to replace high-burnup and/or pebbles with low-fissile content, thereby maintaining the desired excess reactivity in the reactor. The PHSS inserts pebbles at the bottom of the reactor core through the pebble insertion line (PIL). The PHSS also extracts pebbles from the top of the reactor vessel during normal operations with the pebble extraction machine (PEM). Pebbles are examined for burnup and physical damage and are either reinserted into the core or directed to storage.

2.1.2 Coolant

The reactor coolant is a single phase, chemically stable, molten, fluoride salt mixture, 2LiF:BeF₂ (Flibe) enriched in Li-7 (Reference 11) with a high Prandtl number. The coolant acts as a functional containment and provides retention of fission products that escape from any fuel defects.

2.1.3 Reflector

The reactor core itself is formed from graphite reflector blocks that enhance neutron economy and provide the structural pathway for coolant flow through the reactor. The reflector is characterized by multiple axial coolant channels to cool the graphite reflector and small gaps between blocks that result in flow bypassing the reactor core. A primary coolant loop circulates the reactor coolant using a primary salt pump (PSP) and transfers the heat to a heat exchanger for direct rejection to the atmosphere or to a secondary power conversion system.

2.1.4 Reactivity Control and Shutdown Systems

The reactivity shutdown system (RSS) is capable of shutting down the reactor by inserting shutdown elements directly into the packed pebble bed core. Reactivity control in the KP-FHR is accomplished by the reactivity control system (RCS). The RCS inserts and withdrawals control elements outside the pebble bed into the nearby side reflector. For planned power maneuvers of the KP-FHR reactor, only the control elements are used.

2.2 Reactor Core Design

The KP-FHR core contains thousands of randomly packed buoyant pebbles that slowly ascend through the reactor core. The dynamics of the reactor core are characterized by the transition from an initial startup core to an equilibrium core over time. The fuel pebbles may contain natural or depleted uranium and fuel pebbles with enriched uranium less than 20 wt% U-235 to adjust effective enrichment and core reactivity in early startup core operations. Depending on the startup and operational schemes, the core may also contain a fraction of graphite-only moderator pebbles to maintain the desired carbon to heavy metal atom (CHM) ratio. Similar to the moderator to fuel volume ratio in light water reactors, the CHM ratio is used in the KP-FHR to define the neutron moderation conditions (i.e., over-moderated or under-

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moderated). Mixing different pebble types facilitates maintaining the KP-FHR core in under-moderated conditions with desired excess reactivity.

When defining the desired CHM ratio, it is also important to recognize the role of the reactor coolant. Flibe is a moderator but also an absorber due mainly to lithium-6, a natural isotope of lithium (7.59% abundance) with a large thermal absorption cross section. Enriching lithium in Li-7 is required for acceptable core performance (i.e., fuel utilization) and it also ensures negative coolant temperature reactivity feedback.

An increase in the temperature of Flibe leads to a decrease of its density with two competing reactivity feedbacks: a positive feedback due to reduced absorption and a negative feedback due to reduced moderation by the Flibe. The balance of these effects is a function of the CHM ratio (spectrum); therefore, the combined reactivity feedback can be designed to be negative by controlling the CHM ratio. After some period of operation, Li-6 is consumed, and its concentration is lower than in fresh Flibe. Li-6 in Flibe is also produced by (n,α) reactions on Be-9, leading eventually to an equilibrium concentration. Salt impurities that are present in fresh Flibe are also parasitic absorbers in addition to the accumulation of other corrosion material, each of which may have an impact on the coolant reactivity coefficients. The properties and specifications for the reactor coolant are described in Reference 11.

The ability to control the mixture of pebble types in the core allows excess reactivity to be minimized during startup and the transition core (i.e., the core representative of compositional transitions during power ascension). Core reactivity is also controlled by the movement of the control elements. Shutdown elements are also available for insertion for safe shutdown during all core states.

During normal operating conditions, thermal power generated within the fuel is transferred by conduction to the pebble surface. The thermal energy is mainly transferred via convection from the pebble surface by the coolant that flows through the randomly packed bed. At the same time, a smaller portion of the thermal energy is transferred by a mixed regime of conduction and thermal radiation, specifically, pebble-to-pebble heat conduction through a stagnant fluid, pebble-to-pebble conduction, and pebble-to-pebble radiation. A fraction of the total power is deposited in pebbles, Flibe, the reflector blocks, and surrounding structures by radiation heating.

Coolant flows through the vessel downcomer and then through the core, engineered flow channels, and a portion of it bypasses the core through gaps, channels and penetrations within the vessel internal structures. The thermal energy balance within the reactor core determines the temperature distribution within the fuel, moderator, and reflector, and for the coolant that flows through the reactor. The temperature distribution is an input for core reactivity levels, burnup calculations, and power shape.

2.3 Operational Regimes

Each period of core operation is unique in its average core composition. There are four main periods of core operation in the life of the KP-FHR reactor with respect to criticality and composition modeled in the methodology: startup (including low power), power ascension (transition core), approach to equilibrium, and equilibrium. Figure 2-4 provides an illustration of these stages. The methodology for reactor startup modeling considers a critical height approach to an initial fuel load and the approach to criticality. This approach is supported by the methods described in this topical report.

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In the critical height approach, the core is initially loaded with only moderator pebbles. The height of the fueled region of the core is increased by adding a mixture of fuel and graphite pebbles until criticality occurs. Predictions for each step of fuel loading is determined by the 1/M method.

At full power (or at the initial power plateau), the approach to the equilibrium core begins. At startup, the core radionuclide inventory is mostly composed of fresh fuel, and significant burnup on fresh fuel has not yet been accumulated. To compensate for burnup as it accumulates, fresh fuel pebbles are added, and depleted fuel pebbles and natural or depleted uranium pebbles (if present) are removed at a rate that maintains desired core reactivity. After some period of power operation, the isotopic concentration in the core will be largely unchanged. A stable rate of insertion and extraction of fuel will be reached, at which point, the equilibrium core has been reached.

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Figure 2-1 Overview of a KP-FHR Design

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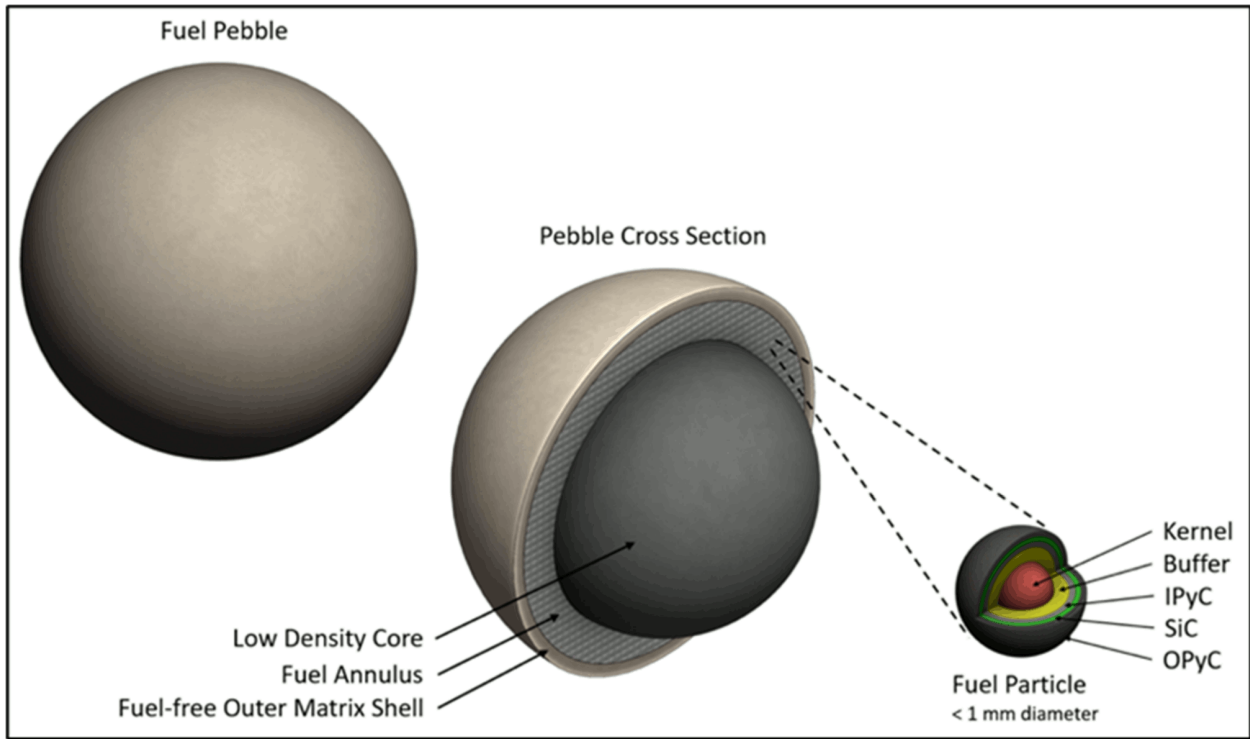
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Figure 2-2 Explicit Serpent 2 Model

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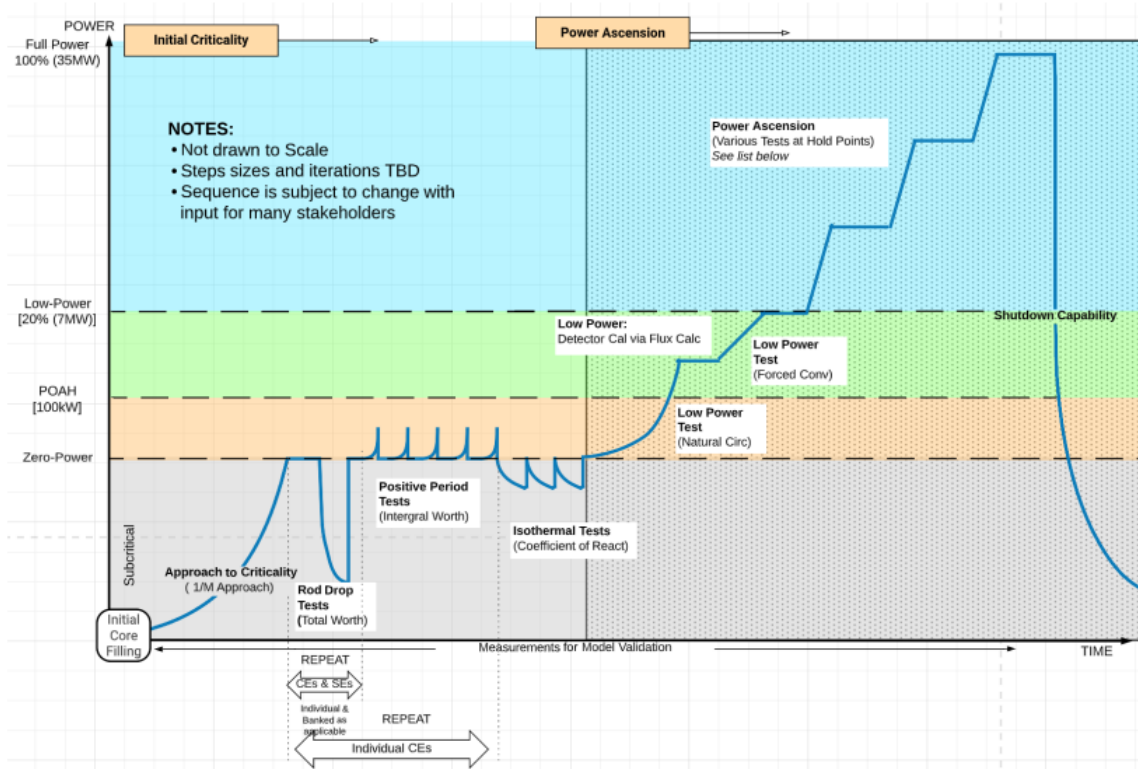
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Figure 2-3 KP-FHR Fuel Pebble and Particle Design



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Figure 2-4 Operational Regimes of a KP-FHR



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3 CORE MODELING PARADIGMS

The KP-FHR core configuration is heterogeneous and non-stationary. The pebble bed evolves from the early startup phase and approaches a statistically steady burnup equilibrium condition. KP-FHR core physical characteristics such as core geometry, heterogeneity, and pebble bed motion require unique modeling approaches. As such, the methods for modeling align very closely with the physical behavior of the core leveraging the modeling tools described in Section 4. The following sections apply to both a KP-FHR power reactor and a KP-FHR test reactor.

3.1 Key KP-FHR Core Steady-State Phenomena

Key phenomena were determined with Phenomena Identification and Ranking Tables (PIRTs), sensitivity studies, industry literature and best practices. These key phenomena and the methods used to determine their importance are discussed in the following sections.

Methods and tools were selected to encompass the physics with the desired level of fidelity at steady state. The PIRT(s) discussed below in Section 3.1.1 and Section 3.1.3 were used for a-posteriori confirmation of the sufficiency of the respective methodologies to capture phenomena relevant to predicting the selected Figures of Merit (FOMs). The KP-FHR internal PIRT was used as a methodology confirmation step along with a literature review. However, it was not used to develop the methodology as conventionally done in the evaluation model development and assessment process (EMDAP).

3.1.1 Key Neutronics Phenomena

A PIRT evaluation was conducted for the KP-FHR steady-state core design. A full review of the existing Georgia Institute of Technology FHR neutronics PIRT (Reference 2), which uses the Advanced High-Temperature Reactor (AHTR) reactor design as the basis, was performed prior to beginning the KP-FHR internal PIRT. The description of Figures of Merit (FOMs) and knowledge level numbering used in the PIRT are as follows:

- FOM 1: Multiplication factor (1: Low , 2: Medium , and 3: High)
- FOM 2: Power distribution (1: Low , 2: Medium , and 3: High)
- Knowledge: Knowledge level (1: Low , 2: Medium , and 3: High)

The KP-FHR internal PIRT, AHTR PIRT, and literature survey were used as a confirmation step for the methodology that revealed adequate fidelity in the best estimate methods to capture the physics of the KP-FHR core. A summary of the KP-FHR internal PIRT results is provided in Appendix C: Neutronics PIRT Results for the KP-FHR.

For phenomena with Medium or Low knowledge level and Medium to High importance, [[

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3.1.2 Key DEM Phenomena

Discrete elements method (DEM) accounts for the random packing and granular motion of pebbles in the core by modeling and predicting the following FOMs:

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- FOM1: Static pebble center locations and packing fraction.
- FOM2: Average pebble track and velocity profiles.

The FOMs above are influenced by the following factors:

- Core shape geometry
- Pebbles contact forces
- Pebbles drag forces
- Pebbles buoyant force

DEM modeling features capture the phenomena listed above. [[

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3.1.3 Key Thermal Hydraulics Phenomena

The core thermal hydraulics scope is to predict steady-state temperature distributions for:

- FOM1: Core materials (Flibe, pebble, TRISO particles)
- FOM2: Reflector graphite

The phenomena for steady state conditions listed below are the same as those in the Kairos Power thermal fluid PIRT and influence the FOMs above. Phenomenon importance in the thermal fluid PIRT was determined for postulated events outside the scope of core thermal-hydraulics applications. Sensitivity analyses for nominal steady state conditions indicate that the following phenomena are of importance:

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The following thermal-hydraulic modeling features allow capture of the phenomena listed above.

[[]]

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3.2 General Modeling Approach

The KP-FHR core modeling paradigm includes discrete elements methods (DEM) (see Section 3.3), neutronics (see Section 3.4), and thermal hydraulics (TH) (see Section 3.5) modules with appropriate coupling between these domains.

DEM simulations generated by STAR-CCM+ (described further in Section 4) are used to explicitly predict the pebble distribution and velocity profile within the core and provide input to the neutronics calculations. The domain of interest for DEM is illustrated in Figure 3-1. DEM simulations use the core shape, the pebbles, and the coolant flow through the core, as input to perform calculations. The core shape, where the pebbles reside, is defined by the reflector structure, which includes the cylindrical section of the core, the upper and lower conic regions, the defueling chute, and the fuel insertion region. Therefore, the DEM modeling paradigm can use this geometry (Figure 2-1 as an example) to provide the boundaries for determining pebble distributions within the core.

The neutronics analyses of the KP-FHR core explicitly account for the double-heterogeneity of TRISO particles and pebbles. The domain of interest for the neutronics analysis is illustrated in Figure 3-1. The explicit neutronics model of the core and reflector structure in Serpent 2 informs the thermal hydraulic model power distribution, which is used to provide material temperature distribution. The neutronics model uses Serpent 2 burnup capabilities to perform depletion calculations. The neutronics model uses a full-core three-dimensional geometry (e.g., Figure 2-1), including the core and pebble bed (which accounts for pebble distributions from DEM simulations), the defueling chute, fueling region, reactivity shutdown system (RSS), reactivity control system (RCS), graphite reflector, reflector penetrations, core barrel, downcomer, and reactor vessel. Like the DEM modeling paradigm, the neutronics modeling paradigm is explicit and uses the geometry of the core.

Thermal hydraulic calculations are performed in three-dimensions using STAR-CCM+. The method includes high-fidelity (i.e., explicit geometrical resolution of domain) and low-order (i.e., porous media) modeling features. The calculational boundaries are illustrated in Figure 3-1 and include the core, defueling chute, fueling region, and the reflector structure with full geometrical resolution (e.g., Figure 2-1), including channels, penetrations, gaps between block columns, gaps between the reflector and core barrel, gaps through vertical and horizontal reflector keys and bypass through the natural circulation flow path to account for bypass flow. The thermal hydraulic modeling of the core region is done through porous media approximation (common in pebble-bed reactor core modeling, e.g., Reference 10 and Reference 12).

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Core analysis calculates quantities of interest for the input into safety analysis, nuclear design, fuel performance, and source term calculations. The domains of interest for modeling are determined before performing core analysis and include the geometric and material boundaries. The transfer of information used as input for applications of this methodology is managed for compliance with appropriate quality assurance requirements. The DEM, neutronics, and thermal hydraulic models are discussed further in the following sections.

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3.3 DEM Modeling Paradigm

The DEM model uses a Lagrangian formulation to describe pebble equations of motion. As described in Section 3.2, STAR-CCM+ is used for both DEM and thermal hydraulic modeling. DEM simulations are performed with a separate module of STAR-CCM+ that uses the pebble geometry, pebble mechanical properties, reflector CAD model (described in Section 2.1.3), reflector mechanical properties, Flibe thermophysical properties, [[]]

DEM Lagrangian formulation uses momentum conservation equations for each pebble, and considers buoyant, drag, and contact forces as shown in Equation 1 below where m_p is the pebble mass and \vec{v}_p is the pebble velocity vector:

$$m_p \frac{d\vec{v}_p}{dt} = \vec{F}_{Drag} + \vec{F}_{Buoyant} + \vec{F}_{Contact} \tag{1}$$

The Flibe flow across the core creates drag forces on pebbles in the packed bed. The influence of fluid drag on pebbles flow is unique to the KP-FHR design in which pebbles are positively buoyant. [[]]

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[[]]

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]] This collected DEM data provides a statistical basis for neutronics and thermal hydraulic calculations to derive the following FOMs:

- FOM 1: Static pebble center locations and packing fraction
- FOM 2: Average pebble tracks and velocity profiles

The above FOMs are used as inputs for neutronics and thermal hydraulic applications. The pebble center locations are used to generate randomly packed pebble bed geometries for neutronics calculations and to provide global packing fraction (porosity) for the thermal hydraulic porous model.

DEM provides detailed information for the transient pebble trajectories inside the core and their statistical average provides input for burnup calculations performed using KPACS (described in Section 4.4.3). The methodology to calculate average pebble trajectories and velocity considers the recirculation of the pebble from core inlet to extraction point at the core outlet. The ZONER tool, described in Section 4.4.3.1, post-processes the DEM simulation outputs to generate spectral zones in the core.

The major model inputs to DEM are:

- Core and pebbles geometry
- Operating conditions (i.e., flow rate)
- Pressure gradient vector field
- Pebble mechanical properties (Young's modulus, Poisson ratio, density)
- Reflector mechanical properties (Young's modulus, Poisson ratio, density)
- Flibe thermophysical properties
- Tribology data for coefficient of friction (COF)

3.4 Neutronics Modeling Paradigm

Explicit neutronics modeling of the KP-FHR core design features (see Section 2) is performed using Serpent 2, which is described further in Section 4.3. Serpent 2 is a continuous-energy Monte Carlo (MC) radiation transport code that provides high-fidelity calculations.

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The model is composed of explicitly modeled pebbles in the core, each with a defined pebble type and composition. TRISO particles are also explicitly modeled within the pebble. As summarized in Section 3.2, the Serpent 2 model includes the graphite reflector with major engineered penetrations and Flibe flow path geometries within the reflector explicitly modeled (see Figure 2-2 based on typical KP-FHR geometry). The reflector region is modeled as a material that homogenizes the Flibe volume fraction and the bulk graphite block. This homogenization accounts for Flibe in the coolant channels (converging and diverging region), gaps between blocks, upper expansion wells, and other minor features. Sections of the reflector with different Flibe fractions can be modeled separately based on the model fidelity required. The neutronic modeling of Flibe regions in the reflector as homogenized material is appropriate because of the small geometric dimensions relative to the neutron mean free path, relatively small Flibe quantity, and the smearing effect of azimuthal symmetry. Features with negligible neutronics impact may be modeled using simplifications including homogenization (e.g., modeling the PEM as a mixture of Flibe and steel based on the volume fractions of each in the design).

The modeling of the RSS and RCS elements include the neutron absorber material and cladding and conservatively neglect the neutronic impact of other structural material. RCS insertion occurs within the reflector channels and is modeled by lowering the RCS bank into these channels, while removing Flibe from the areas overlapping with the RCS rods. The RSS bank is inserted into the pebble bed using the "cut pebble bed" method. This method involves cutting the pebbles and removing Flibe from the regions overlapping with the RSS rods, which reduces the amount of fuel and Flibe in the core when the rods are inserted.

The reactor is modeled neutronicly within the boundaries described in Section 3.2 (i.e., Figure 3-1), and is used to model various core states or material compositions. Material compositions are defined using available specifications, best-available estimates, and conservative estimates. Impacts on material composition uncertainty are considered and quantified in Section 5.2.3. Temperature feedback effects in the reactor are analyzed using KPATH, described in Section 4.4.4.

The nuclear data files used by Serpent 2 are produced by NJOY21 with inputs defined by KACEGEN, which is described further in Section 4.4.2. The outputs from neutronics modeling are summarized in Section 3.6. The calculational methodologies for deriving the FOMs are described in Section 6.1. The KP-FHR reactor is modeled with the general features described in Section 2 and due to the reactor design, the Serpent 2 model can be modified or expanded as necessary to represent different neutronics phenomena and calculate the figures of merit (FOMs) in Section 3.6.

3.5 Thermal Hydraulics Modeling Paradigm

Table 3-1 provides an example of general thermal hydraulic parameters of KP-FHR test reactors during normal steady state operations. The FOMs for the thermal hydraulic analyses of the KP-FHR are:

- 1) Core materials temperature distribution
- 2) The graphite reflector temperature distribution

Thermal hydraulic analyses are used to update material cross section temperatures for reactivity calculations and to provide fuel and reflector temperature distributions (with associated uncertainties) to ensure operation within steady-state qualification limits.

Core thermal hydraulics modeling is applicable to steady-state KP-FHR core operations, assuming the modeled conditions are within the range of applicability in the constitutive models.

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A local thermal non-equilibrium (LTNE) porous media (PM) model (References 10, 12, and 17) is used to model the packed bed core mass, momentum, and energy transport. The model provides a macroscopic volume averaged representation of pebbles to determine fluid flow and heat transfer. To close mathematical model formulations, the methodology uses correlations for pressure drop and heat transfer.

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Figure 3-2 summarizes the modeling boundary for the core thermal hydraulics methodology, which provides a schematic representation of the integral KP-FHR TH model. The model consists of a porous media core region [[]]. The porous media core models the heat transfer between Flibe and solid pebble phases, including graphite pebbles and fuel pebbles. [[

]]

The neutronics model provides power density fields for core and reflector regions as an input to the thermal-hydraulics model. The FHR power shape is not sensitive to temperature feedback, therefore a one-way coupling is used as the baseline modality of data transfer for thermal hydraulics (Reference 18). An example of power shape sensitivity to temperature distributions can be found in the generic-FHR benchmark in Reference 18. [[

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two-way coupling calculation using the KPATH tool is described in Section 4.4.4 and has been performed to validate this assumption.

3.5.1 Porous Media Model

This section describes the three correlations that are used for packed bed porous media momentum and heat transfer modeling. The applicability of correlations is assessed for a range of operations where the

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methodology applies, as described in Section 5.3.8. Table 3-3 provides a summary of the non-dimensional numbers used in these correlations and their definitions.

3.5.1.1 Pressure Drop Correlation

The KTA correlation (Reference 7) is used to model the pressure drop ΔP across the packed bed as expressed in Equations 5 and 6:

$$\frac{\Delta P}{\Delta H} = \Psi \frac{1 - \epsilon}{\epsilon^3} \frac{1}{D_p} \frac{1}{2\rho_f} \left(\frac{\dot{m}}{A}\right)^2 \quad (5)$$

$$\Psi = \frac{320}{\left(\frac{Re_p}{1 - \epsilon}\right)} + \frac{6}{\left(\frac{Re_p}{1 - \epsilon}\right)^{0.1}} \quad (6)$$

where ΔH is the height of the bed and ϵ is the porosity of the bed, D_p the pebble diameter, ρ_f is Flibe density, \dot{m} is the mass flow rate, A is the bed flow area (no pebbles), and Re_p is the Flibe Reynolds number based on pebble diameter. KTA correlation implementation in the porous media model is discussed further in Section 3.5.1.3.

[[Reference 65 demonstrated that the [[]] conducted at Texas A&M University. [[]]

]]

Additionally, [[]] with Argonne National Laboratory (ANL) LES NekRS (Reference 66) [[]]

]]

3.5.1.2 Heat Transfer Correlation

The Wakao correlation (Reference 8) as shown in Equation 7, is used to model the fluid-to-pebble heat transfer in the packed bed:

$$Nu_w = 2 + 1.1 Pr^{1/3} Re_p^{0.6} \quad (7)$$

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]] Section 3.5.1.3 describes how these correlations are implemented in the porous media model formulation.

[[Reference 66 demonstrated that the [[

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3.5.1.3 LTNE Porous Media Formulation

[[

]] The local thermal non-equilibrium (LTNE) porous media formulation is used for the core region only. LTNE is used to capture the convective heat transfer between pebbles and Flibe. In this formulation, the pebbles and Flibe have different temperatures. Flibe is referred to as “fluid-phase” and pebbles referred to as “solid-phases.” The baseline porous model solid-phase can include all the fuel pebbles passes (burnup groups) and moderator pebbles. The model has the fidelity to provide core temperature distributions for each solid phase. The baseline porous methodology can be adapted to suit specific applications (e.g., use of only one average solid phase, biasing results towards higher temperatures). The LTNE formulation for mass, momentum, and energy transport for Flibe and pebbles (i.e., fuel pebbles and graphite pebbles) are presented below. The implementation of pressure drop and heat transfer correlations are also provided below. Examples of applications of LTNE are found in References 10 and 12. The LTNE porous media model also models power distribution between core materials as shown in Table 3-5. The LTNE porous media model provides the surface average temperature of pebbles. To model the temperatures within the pebbles and TRISO particles, a 1D thermal conduction model is implemented and uses the pebble surface temperature from porous media as a boundary condition as shown in Figure 3-3.

The geometrical characterization of the packed bed is described by three inputs. Porosity ϵ , represented in Equation 9, solid-phase volume fraction α_{si} , represented in Equation 10, and interaction area density α_{wi} , represented in Equation 11. The latter represents the i -th solid-phase surface area per unit volume inside the bed.

$$\epsilon = \frac{V_{fluid}}{V_{bed}} \quad (9)$$

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$$\alpha_{si} = \frac{V_{solid,i}}{\sum_i V_{solid,i}} \quad (10)$$

$$a_{wi} = \alpha_{si} \frac{6(1 - \epsilon)}{D_p} \quad (11)$$

Where, $V_{solid,i}$ is the volume of the i-th solid porous phase (e.g., fuel pebbles and graphite pebbles), ϵ is the core porosity and D_p is the pebble diameter.

[[

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An example of derivation for the LTNE formulation can be found in Reference 12, and it is based on the application of a volume average operator to the mass, momentum, and energy equations for fluid and solid porous media phases.

The steady-state formulation includes the following transport equations:

- Flibe mass conservation
- Flibe momentum transport
- Flibe energy transport
- Pebbles energy transport

The transport equations are based on the physical velocity formulation that accounts for the increase of fluid velocity when it enters the porous medium. The relation between physical velocity and superficial velocity (the Dupuit-Forchheimer relation in Reference 17) is shown in Equation 12:

$$\vec{v}_s = \epsilon \vec{v} \quad (12)$$

Where \vec{v}_s is the superficial velocity vector, \vec{v} is the physical velocity vector and ϵ is the porosity.

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Flibe Mass Conservation Equation

The Flibe-phase mass conservation equation is represented by Equation 13, where ρ_f is the Flibe density:

$$\nabla \cdot (\epsilon \rho_f \vec{v}) = 0 \quad (13)$$

Flibe Momentum Transport Equation and KTA Correlation Implementation

Flibe-phase momentum transport is represented by Equations 14 and 15:

$$\nabla \cdot (\epsilon \rho_f \vec{v} \vec{v}) = -\epsilon \nabla P + \nabla \cdot (\epsilon \bar{\bar{T}}) - \epsilon \vec{f}_p + \epsilon \vec{f}_b \quad (14)$$

$$\vec{f}_b = \rho_f \vec{g} \quad (15)$$

where, P is the pressure, $\bar{\bar{T}}$ is the velocity shear stress tensor, f_p is the porous resistance force, f_b is the body force and μ_f is the fluid viscosity. [[
]]

The porous resistance force, shown in Equation 16, is expressed in the form of Dupuit-Darcy-Forchheimer (References 12, 17, 19, and 20) to account for linear and quadratic superficial velocity components of pressure losses typical of porous media (Reference 17):

$$\vec{f}_p = P_v \vec{v}_s + P_i |\vec{v}_s| \vec{v}_s \quad (16)$$

where P_v is the viscous porous isotropic resistance tensor and P_i is the inertial porous isotropic resistance tensor.

The KTA correlation is implemented by re-arranging its original formulation (Reference 7) in Dupuit-Darcy-Forchheimer form as isotropic resistance tensors, Equations 17 and 18:

$$P_v = 160 \frac{(1 - \epsilon)^2 \mu_f}{\epsilon^3 D_p^2} \quad (17)$$

$$P_i = 3 \frac{(1 - \epsilon) \rho_f}{\epsilon^3 D_p} \left(\frac{Re_p}{1 - \epsilon} \right)^{-0.1} \quad (18)$$

The KTA porous resistance force can be implemented as described in Equation 19:

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$$\vec{f}_p = 160 \frac{(1-\epsilon)^2 \mu_f}{\epsilon^3 d_p^2} \vec{v}_s + 3 \frac{(1-\epsilon) \rho_f}{\epsilon^3 d_p} \left(\frac{Re_p}{1-\epsilon} \right)^{-0.1} \vec{v}_s |\vec{v}_s| = \nabla P_d \quad (19)$$

In Equation 19, ρ_f is the density of Flibe, \vec{v}_s is the Flibe superficial velocity vector, ϵ is the average bed porosity, d_p is the pebble diameter, μ_f is the Flibe viscosity, and Re_p is the Reynolds number based on pebble diameter and superficial velocity. Equation 19 also models the pressure gradient vector ∇P_d used in DEM models to model pebble drag force. The pressure gradient is extracted and implemented as pebble volumetric body force. This has been described in Section 3.5.42

Flibe Energy Transport Equation and Wakao Correlation Implementation

The Flibe-phase energy transport equation is represented in Equation 20:

$$\nabla \cdot (\epsilon \rho_f H_f \vec{v}) = -\nabla \cdot (\epsilon \vec{q}_f) + \nabla \cdot (\epsilon \vec{T} \vec{v}) + \sum_{\substack{\text{solid} \\ \text{phases}}} a_{wi} HTC_{si} (T_f - T_{si}) + \epsilon \vec{f}_b \cdot \vec{v} + S_f^e \quad (20)$$

where the heat flux is defined as:

$$\vec{q}_f = -k_{f,eff} \nabla T_f \quad (21)$$

And where H_f is the total enthalpy of the fluid, $k_{f,eff}$ is the effective fluid thermal conductivity, HTC_{si} is the heat transfer coefficient for the fluid-solid phase for the i -th solid phase, S_f^e is the volumetric energy source term (power deposited in Flibe), T_f is the local mass flow averaged temperature of the fluid, and T_{si} is the local average surface temperature of the i -th solid phase. The effective fluid-phase thermal conductivity is an input to this methodology.

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The Wakao correlation (References 8 and 21) is implemented by expressing the energy equation in terms of heat transfer coefficient as shown in Equation 22:

$$HTC_{si} = Nu_w \frac{k_f}{D_p} = \frac{k_f}{D_p} \left(2 + 1.1 Pr^{1/3} Re_p^{0.6} \right) \quad (22)$$

The correlation is used to express heat transfer between the Flibe and pebbles (i.e., fuel pebbles and moderator pebbles).

Pebbles Energy Transport Equation

The pebble-phase energy transport equation is provided in Equation 23:

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$$-\nabla[(1 - \epsilon)\alpha_{si}\vec{q}_{si}''] + a_{wi} \text{HTC}_{si} (T_{si} - T_f) + S_{si}^e = 0 \quad (23)$$

where the heat flux is defined as:

$$\vec{q}_{si}'' = -k_{si,eff}\nabla T_{si} \quad (24)$$

And where $k_{si,eff}$ is the i-th effective solid thermal conductivity and S_{si}^e is the volumetric energy source term for i-th solid phase (power deposition in the i-th solid phase). [[

]]

The effective solid-phase thermal conductivity is an input to this methodology. [[

]]

The effective solid thermal conductivity refers to the fuel pebbles layers thermal homogenization. Two equations of this form are used, one for fuel-pebbles and one for graphite pebbles. The model does not account for fuel-to-moderator pebble heat transfer due to negligible radiation and contact heat transfer.

Core-Reflector Boundary Conditions [[

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Pebble and TRISO 1-D Heat Transfer Model

Power is expressed in the form of power density distributions from the coupled neutronics model and is defined in the equations as volumetric source term in the corresponding porous-phase energy formulation. [[

]]

The LTNE model provides the fuel pebble outer surface temperature and moderator pebble outer surface temperature distributions. The fuel pebble surface temperature field is used as a boundary condition for the 1D pebble model. The one-dimensional TRISO model is connected to a reference temperature within the pebble fuel layer, as described in the schematic Figure 3-6.

Pebble power distribution field, \dot{Q}_{PB} , shown in Equation 28, is evaluated from the power density pebble peaking factor distribution shown in Equation 29:

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The TRISO one-dimensional model uses a reference temperature T_{ref} of the pebble fuel layer [[
]] as an outer boundary condition for T_{OPyC} as shown in Figure 3-4. TRISO power shown in Equation 34 is evaluated from the corresponding pebble power and a constant TRISO peaking factor. The fuel kernel is the only region modeled with a power source. PF_{TRISO} in Equation 34 represents the TRISO power peaking within the pebble fuel layer. [[

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3.5.2 Reflector Modeling

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3.5.3 Thermal Hydraulic Model Use

The core thermal-hydraulics methodology uses [[

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3.5.4 [[]] Computational Model Design and Implementation Example

This section describes the general CFD [[]] modeling approach used for the reflector and to generate the validation data for the porous media modeling used in the core. A description and examples are provided on the boundary conditions, mesh design approach, turbulence and wall modeling, and numerical solution verification. The term “family of KP-FHR designs” used in CFD is defined as a group of designs having similar geometry, power density and flow rates. For example, Hermes is considered its own family while Hermes 2 and KP-FHR commercial are a separate family from Hermes.

3.5.4.1 Geometric modeling approach and simplifications

The general approach to geometrical modeling is to minimize the need for geometrical simplifications. CFD modeling provides the capability of discretizing and resolving geometrical features from the actual design. The level of detail is adapted to the scope and objective of the simulation. In cases of geometrical simplification, sensitivities studies are performed to evaluate the effects on FOMs. Geometrical simplification effects are also included if necessary to the overall FOMs uncertainty characterization.

In a typical core TH integral model, the following design details are captured, see Figure 3-7 for graphical example:

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- Fluid geometry
 - Core geometry
 - Downcomer
 - Natural circulation flow path
 - Lower Plenum
 - Core Inlet/Outlet Channels
 - Reflector blocks Gaps (e.g. Spoke, Rim gaps)
 - Upper Plenum/Pump-Well
 - Bypass Mitigation Features (Keys)
 - Reflector Penetrations and Gaps
 - Reflector Cooling Channels
- Solid geometry
 - Reflector
 - Vessel
 - Barrel
 - RCS/RSS
 - PEM

The level of geometrical resolved detail is dependent on the design and scope of the simulation. The listed geometry above is an example and dependent on the scope of the problem being analyzed. In the example above, steel structures are included (although metallics temperatures are outside the scope of this topical report) as they provide detailed geometrical boundaries to the main coolant flow path.

3.5.4.2 Material properties

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3.5.4.3 Mesh Design Approach

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3.5.4.4 Boundary Conditions and Energy Source Terms

Examples of domain boundary conditions are provided in Tables 3-6 through Table 3-10 and referenced in Figure 3-9. The types and location of boundary locations are strictly dependent on the design and scope of the CFD simulation and are subject to change. In this example, domain symmetry is utilized, together with adiabatic boundaries to neglect thermal leakage. The latter, for example, could be changed to convective, radiative, or mixed type of boundary if more realistic thermal leakage modeling is necessary for the scope of simulation.

In addition to boundary conditions for momentum and energy, the model requires energy source terms (power density fields) in both the core region (in-core) and the outer core region (ex-core). This is necessary to account for fission and radiation heating across the domain, as described in Section 3.5.1.3 (porous media) and Section 3.5.2 (reflector model). As explained in Section 6.1.4, the power density fields are extracted from the Serpent 2 model and interpolated onto the CFD computational mesh. As part of solution verification, the total energy conservation is assessed in the CFD model to ensure conservation throughout the domain.

Examples of energy source terms are included in Table 3-11 for the core region (in-core) and Table 3-12 for the outer core region (ex-core).

3.5.4.5 Turbulence and Wall Modeling Approach

The thermal-hydraulics methodology described in this report makes use of [[

]] For every family of KP-

FHR designs the following assessment process is based on general best-practices and sensitivity studies:

1. Quantitative assessment of turbulence and wall models on FOMs via sensitivity analyses [[

]]

2. Turbulence sensitivities results are compared with other sources of uncertainties (e.g., heating, boundary conditions, inputs).

3. Effects of [[are assessed for the turbulence models as part of the model sensitivities described in point 1.

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4. Wall Y^+ and T^+ are always monitored on selected surfaces to make sure thermal and velocity boundary layers are sufficiently resolved and meet the general requirements for their applicability (e.g., $Y^+ < 1$ for low- Y^+ models, $Y^+ > 30$ if wall functions are used)
5. Assessment of turbulent viscosity ratio in key regions of the domain to quantify [[
]] impact on analyses and FOMs.

KP-FHR designs are characterized by low coolant flow rates throughout the domain due to Flibe’s high volumetric heat capacity. For instance, the core Reynolds number for Kairos Power test reactor designs can range from [[

]] In comparison, pebble bed HTGR designs have core Reynolds number that can span from 15,000-25,000 (e.g., THTR reactor design). [[

]]

3.5.4.6 Numerical Solution

Second order numerical schemes are primarily used for the steady-state CFD [[
]] modeling. Numerical solution verification includes iterative convergence assessment. This includes monitoring of the following local and global quantities for mass, momentum, and energy equations:

- Global monitoring:
 - Mass balance
 - Total power
 - Vessel outlet temperature
- Local monitoring:
 - Velocity, temperature at selected point locations
 - Y^+ monitoring at selected surfaces locations
- FOM (global) monitoring:
 - Pressure drops
 - Nusselt number at selected locations
 - Mass flow rates at selected fluid interfaces or boundaries
- Normalized Iteration residuals

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3.6 Summary of Computational Outputs

Using the modeling paradigms and boundaries described above for core analysis, the following calculated output quantities for different core states are determined and used in core design as noted below:

- [[

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The following output parameters from the core design and analysis methodology is used as input into safety analysis calculations:

- Reactivity coefficients
- Kinetics parameters
- Shutdown margin

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- Rod worth
- Axial and radial power shape
- Pebble peaking factor (fuel performance)

Conservative uncertainties are applied to these parameters per the methodology provided in Section 5. Further discussion of the application of these output parameters is provided in Section 6.

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Table 3-1 Example of Thermal Hydraulic Parameters of KP-FHR Test Reactors³

TH Features	Hermes	Hermes 2.0
Thermal Power [MW]	[[]]	35
Vessel inlet/outlet temperatures [°C]	550-650	550-650
Pebble Power [W]	[[]]	[[]]
Peak TRISO SiC temperatures [°C]	770	850
Peak TRISO kernel temperatures [°C]	820	1000
Reflector peak temperatures [°C]	[[]]	[[]]
Flibe Pr	[[]]	[[]]
Core Re_p^1	[[]]	[[]]
Core inlet/outlet channels Re_{Dh}^2	[[]]	[[]]
Reflector cooling channels Re_{Dh}^2	[[]]	[[]]
Reflector blocks gaps Re_{Dh}^2	[[]]	[[]]
Core porosity	40-42%	40-42%

Notes

1. Re_p is the Reynolds number based on pebble diameter and superficial velocity.
2. Re_{Dh} is the Reynolds number based on hydraulic diameter and average velocity.
3. The design information reflected in the examples is to illustrate the methodology and not convey final design information.

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Table 3-2 Model Paradigm Summary

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Table 3-3 Non-dimensional Numbers Definition

Number	Definition	Variables	Nomenclature
Reynolds	$Re = \frac{\rho v_s D_p}{\mu} = \frac{\dot{m} D_p}{\mu A}$	ρ	Density (kg/m ³)
		v_s	Superficial velocity (m/s)
		D_p	Pebble diameter (m)
		μ	Dynamic viscosity (Pa-s)
		A	Flow area (m ²)
		\dot{m}	Mass flow rate (kg/s)
Prandtl	$Pr = \frac{\mu C_p}{k}$	C_p	Specific heat capacity (kJ/kg K)
		k	Thermal conductivity (W/m K)
Nusselt	$Nu = \frac{HTC D_p}{k}$	HTC	Heat transfer coefficient (W/m ² K)
Richardson	$Ri = \frac{g\beta\Delta T D_p}{v_s^2}$	g	Gravity acceleration (m/s ²)
		β	Thermal expansion coefficient (1/K)
		ΔT	Reference temperature difference (K)

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Table 3-4 LTNE Boundary Conditions for the Core and Reflector

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Table 3-5 Power Distributions FOM

Region	Power	Description
IN Core	Fuel	Power deposited in fuel pebbles
	Fuel Pass X	Power deposited in fuel pebbles of a given Pass X
	Graphite	Power deposited in graphite pebbles
	Flibe	Power deposited in Flibe
EX Core	Graphite	Power deposited in reflector
	Flibe	Power deposited in Flibe
	Steel	Power deposited in steel structures

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Table 3-6 Flibe momentum boundaries type

Flibe - Momentum			
Boundary	Region Surface	Type	Rationale/Description
Inlet	Downcomer	Mass Flow Rate	[[]]
Outlet	Pump-Well	Pressure Outlet	[[]]
Free Surface	Top Domain	Slip	[[]]
Symmetry Plane	Symmetry	Symmetry	[[]]

Table 3-7 Flibe energy boundaries type

Flibe - Energy			
Boundary	Region Surface	Type	Rationale
Inlet	Downcomer	Temperature	[[]]
Outlet	Pump-Well	Zero Gradient	[[]]
Free Surface	Top Domain	Adiabatic	[[]]
Symmetry Plane	Symmetry	Symmetry	[[]]

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Table 3-8 Reflector energy boundaries type

Reflector - Energy		
Boundary	Type	Rationale
Top Surface	Adiabatic	[[]]
Symmetry Plane	Symmetry	[[]]

Table 3-9 Vessel energy boundaries type

Vessel - Energy		
Boundary	Type	Rationale
Top Surface	Adiabatic	[[]]
Symmetry Plane	Symmetry	[[]]
Outer Surface	Adiabatic	[[]]

Table 3-10 Other solid-domain energy boundaries type

Core Barrel/PEM/RSS/RCS - Energy		
Boundary	Type	Rationale
Top Surface	Adiabatic	[[]]

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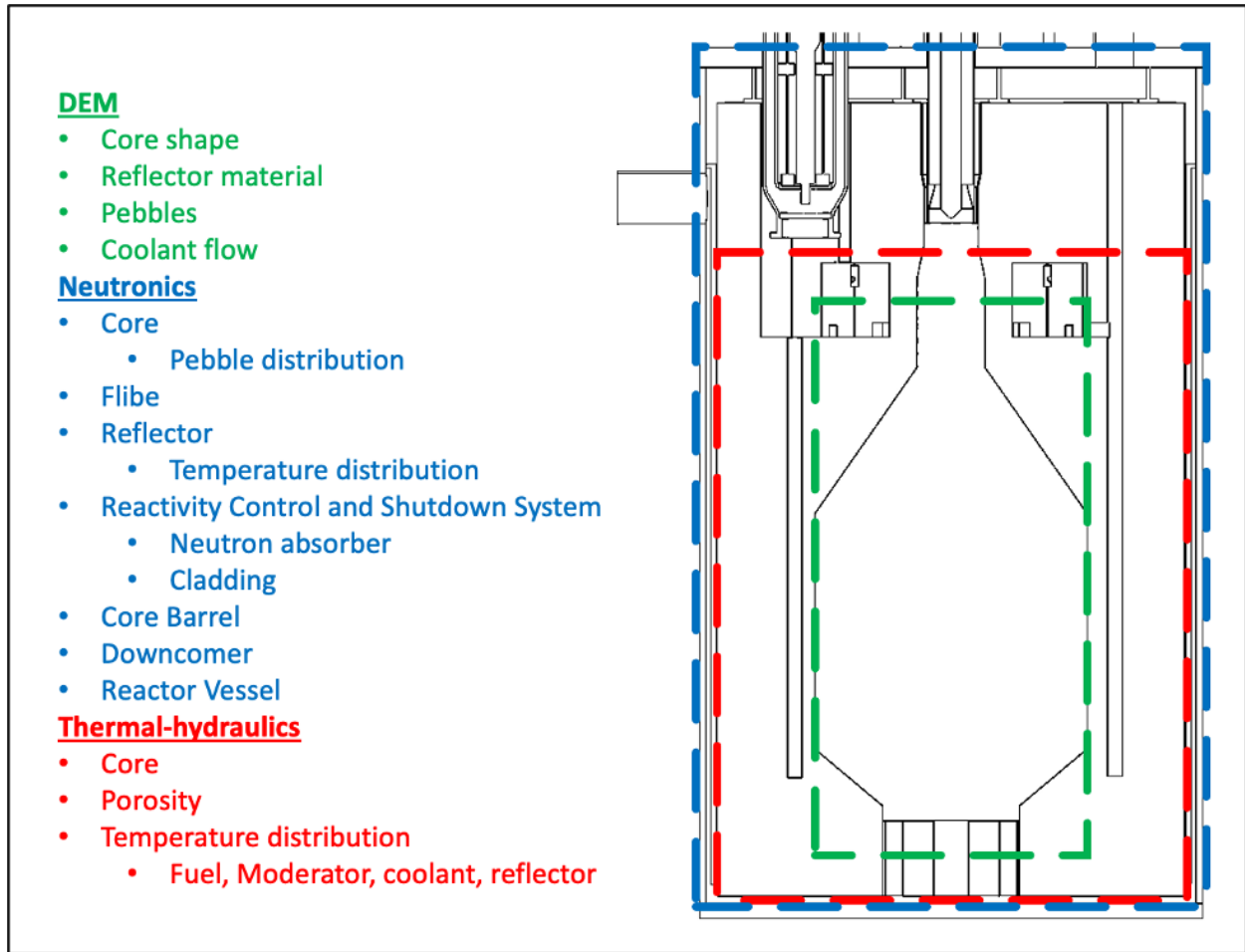
Table 3-11 Energy source (In-Core)

Energy Source (Power Density W/m ³)		
Region	Power	Description
In-Core	Fuel Pebbles	[[]]
	Graphite Pebbles	[[]]
	Flibe	[[]]

Table 3-12 Energy source (Ex-Core)

Energy Source (Power Density W/m ³)		
Region	Power	Description
Ex-Core	Graphite	[[]]
	Flibe	[[]]
	Steel	[[]]

Figure 3-1 Core Modeling Domains



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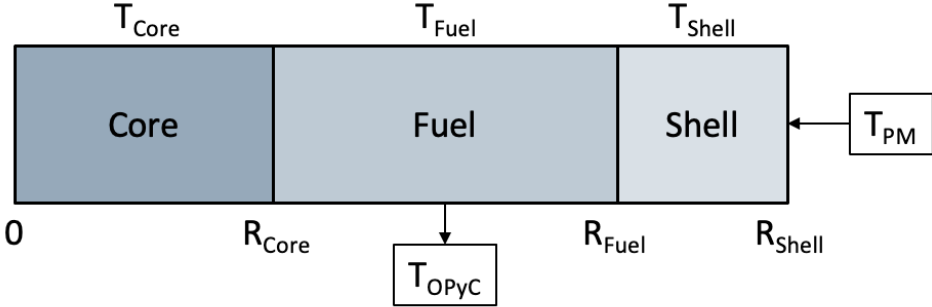
Figure 3-2 Integral TH modeling domains framework

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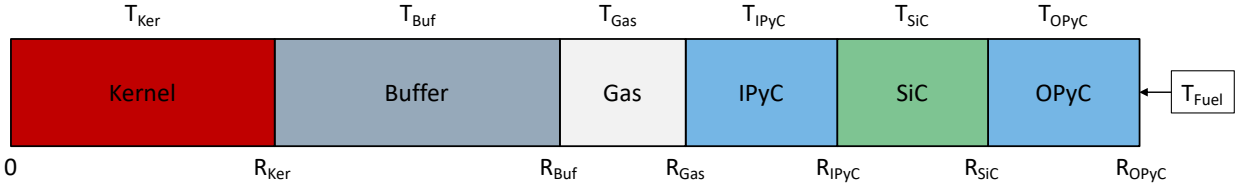
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Figure 3-3 Pebble Boundaries Used in the Porous Media Model



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Figure 3-4 TRISO Boundaries Used in the Porous Media Model



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Figure 3-5 Connection between porous media model solid phase pebbles surface temperature and 1 D conduction models

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Figure 3-6 Fuel pebble and moderator pebble 1-D conduction model nodalization

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Figure 3-7 **Examples of lateral and top views of solid domain (left) and fluid domain (right)**

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Figure 3-8 Example of conformal polyhedral mesh for Hermes design

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Figure 3-9 Example of computational domain surfaces boundaries for fluid and solid domain

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Figure 3-10 Example of turbulence model sensitivity analysis compared with other sources of uncertainty

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4 MODELING TOOLS

The software discussed in this report is developed and maintained under the Kairos Power Quality Assurance (QA) program, which is based on industry standard methods.

Software verification aims to ensure that the discretized model is an accurate representation of the continuous mathematical model, and that there are no user-defined code errors. Verification is performed on STAR-CCM+, Serpent 2, and the wrapper codes.

Validation is the process of determining the degree to which a mathematical model is an accurate representation of the real world from the perspective of its intended use. This is done by comparing the model outputs (FOMs) with experimental measurements and/or higher fidelity numerical results. Validation is only performed for STAR-CCM+ and Serpent 2 because these are the underlying software codes that are used to perform the reactor physics and thermal hydraulic analysis. The wrapper codes are only used to process and exchange data between STAR-CCM+ and Serpent 2 and the libraries. The verification process consists of software benchmarking and numerical solution verification, and is described further in Section 5.

The section below introduces the software tools as they relate to the core design and analysis methodology DEM, neutronic and thermal hydraulic models, data exchange between codes, and the FOMs generated for a KP-FHR test reactor and power reactor. More detailed code descriptions are provided for STAR-CCM+ and Serpent 2 as they are the main drivers for the methodology. The other codes are used to process and transfer information between the codes and do not explicitly model core physics.

4.1 Process Flow

The KP-FHR steady state and pseudo-steady state (i.e., in the case of pebble motion) core design modeling workflow and data exchange consists of different degrees of coupling between DEM, neutronics, and TH modules. Figure 4-1 presents a graphical summary of the data flow and processing of the core modeling paradigm.

The first step in the methodology is to generate the randomly packed pebble bed core geometry. DEM modeling provides the coordinates of pebbles. TRISO particle distributions are randomized within the fuel region of pebbles. The DEM model provides all the data necessary to account for pebble residence time profiles. The neutronics module then uses the geometrical and pebble bed motion data provided by the DEM to build the core geometry and spectral zone discretization for criticality and burnup calculations. The thermal hydraulic module uses the pebble packing fraction from DEM, material and geometry inputs and operating conditions to inform the TH model. The neutronics and TH models are coupled (one-way and two-way) by power and temperature distributions to provide thermal hydraulic feedback.

4.2 STAR-CCM+

STAR-CCM+ is used for DEM and TH modeling. STAR-CCM+ is a multi-physics computational fluid dynamics simulation code. STAR-CCM+ models fluid flow and heat transfer, as well as heat transfer in solids, for complex three-dimensional geometries. It solves the Navier-Stokes equations discretized using the finite volume approach for steady state and time-dependent problems. STAR-CCM+ is used to represent solid, liquid, and porous media. Heat may be transferred via conduction, and convection. As

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described in Section 5, the verification, validation (V&V) and uncertainty quantification methodology reduces and controls the sources of error and uncertainty between STAR-CCM+ models used in core design and their predictive capabilities. The V&V methods for DEM and TH are provided in Section 5. The TH V&V methodology focuses on the prediction of the core material temperatures (reflector, fuel, moderator, and coolant) whereas the DEM methodology focuses on the pebble center locations and their residence time profiles within the core. [[

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4.3 Serpent 2

Serpent 2 (Reference 39) is the main neutronics tool for reactor core design, input into safety analysis, fuel performance, and source term calculations. Serpent 2 has been extensively used across academia and industry and has been validated against various benchmarks. It is used in KP-FHR nuclear design for a variety of calculations, including multiplication factor, control and shutdown element worths, reactivity coefficients, power distribution, kinetics parameters, nuclear heating, burnup calculations, and activation analysis (see Section 3.6). Kairos Power made no modifications to the base Serpent 2 code.

Serpent 2.2.1 is used for the calculations provided within this topical report. This methodology may be applied using future versions of Serpent 2, and successive code versions will be managed under the Kairos Power software quality assurance program, discussed in Section 4.6. The applicability of future code versions will be evaluated through computer test cases to identify major changes, abnormalities, or deficiencies and by performing code-to-code benchmarking as shown in Section 5.2.

4.3.1 Geometry and Particle Tracking

Serpent 2 uses a three-dimensional constructive solid geometry (CSG) model for its basic geometry routine, which is a common choice in Monte Carlo particle transport codes. The model is built using quadratic and derived surface types, forming two- or three-dimensional cells. It allows for hierarchical structuring through universes, transformations, and repeated structures like square and hexagonal lattices. A stochastic geometry model is available for handling randomly distributed fuel particle for TRISO and pebble modeling, with individual fuel particle/pebble coordinates read from a separate input file (generated using the DEM model as described in Section 3.3). Particle tracking in Serpent 2 combines conventional ray-tracing surface tracking with the rejection sampling-based delta-tracking method. Delta-tracking is useful in reactor calculations due to the typically long neutron mean-free-path compared to spatial dimensions, which results in a significant speed-up. In cases of localized heavy absorbers causing efficiency issues, the transport routine switches to surface-tracking. Delta-tracking in Serpent 2 requires that collision estimators be used for flux tallies and reaction rate integrals. The efficiency of delta-tracking in geometries with many surfaces enables Serpent 2 to model the KP-FHR with explicit pebble and TRISO particle definitions instead of approximations.

4.3.2 Interactions

Serpent 2 uses continuous-energy ACE or A Compact ENDF format cross section libraries to read neutron interaction data, serving as the foundation for the laws of physics in transport simulations. This format was originally developed for MCNP and now being leveraged by other MC tools such as OpenMC and Geant4. The software supports the use of separate thermal scattering data for moderator materials, includes probability table sampling in the unresolved resonance region, and enables the adjustment of nuclide temperatures through a built-in Doppler-broadening preprocessor routine. Additionally, the

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Doppler-broadening Rejection Correction (DBRC) method is employed to account for the temperature dependence of resonant scattering kernels.

Serpent 2 allows reconstruction of all cross sections using a unified energy grid. There are two benefits to using this option. First, the energy grid search required for interpolating microscopic cross sections is achieved only once for each neutron scattering event to a new energy, which reduces calculation time. Secondly, pre-calculation of material-wise macroscopic cross-sections minimize the need for summation over constituent nuclides during transport simulations, which enhances computational efficiency. Photon transport capabilities, including fission energy deposition distribution are added functionalities in Serpent 2. These capabilities are also used in modeling KP-FHRs and demonstrated in the validation, verification, and uncertainty analysis process and described in Section 5.2.

4.3.3 Burnup

Burnup calculations in Serpent 2 rely on internal subroutines, independent of external depletion solvers. The software automatically calculates decay and transmutation paths and selects nuclides for calculation without input from the user. The software gets radioactive decay data, energy-dependent fission yields, and isomeric branching ratios for neutron reactions from ENDF format data libraries.

The evolution of material undergoing depletion or burnup radioactive materials under neutron flux is described by the Bateman equation. The differential equation describing the rate of change for a single nuclide is given by the following equation (Reference 40):

$$\frac{dN_j}{dt} = \sum_{i \neq j} \lambda_{ij} N_i + \sum_{i \neq j} \Phi \sigma_{ij} N_i - \lambda_j N_j - \Phi \sigma_j N_j \quad (39)$$

where N_j is the concentration of nuclide j , λ_{ij} is the decay constant for decay mode of nuclide i producing nuclide j , Φ is the neutron flux, σ_{ij} is the cross-section for neutron-induced production of nuclide j by nuclide i , λ_j is the decay constant of nuclide j , and σ_j is the removal cross-section of nuclide j . The flux and cross-section terms expressed in Equation 39 are energy-collapsed 1-group values.

During transport simulations, one-group transmutation cross sections, essential for the Bateman depletion equations, are calculated in real time. This method of spectrum collapsing is also used in other Monte Carlo burnup calculation codes. The irradiation history is split into different intervals with different normalizations based on power, power density, total flux, fission, or source rate. Depletion steps, for time or burnup steps, can be carried out using conventional Euler and predictor–corrector methods with linear interpolation, or more advanced techniques involving linear and quadratic interpolation and extrapolation.

In Serpent 2, the solution to the Bateman depletion equations is determined through the Chebyshev Rational Approximation Method (CRAM) matrix exponential method (Reference 41). This method provides solutions for complete nuclide systems without approximations for short-lived isotopes and limitations on step length. The method also has minimal numerical precision issues, and its accuracy and efficiency has been demonstrated to be suitable for burnup calculations which are characterized by a high number of nuclides and depletion zones. A comprehensive study of this is demonstrated by Isotalo and Aarnio in Reference 42.

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4.4 Wrapper Codes

The following wrapper codes are used to transfer information between the neutronic and thermal-hydraulic codes. Wrapper codes perform data transfer and do not contain physical models that need to be validated, however their integral functions are verified.

4.4.1 HEEDS

The HEEDS code is a design and analyses software that provides tools that encompass sensitivity analyses. HEEDS couples with STAR-CCM+ and is used to perform input sensitivity and uncertainty analyses. The sensitivity analysis can be configured with a baseline STAR-CCM+ simulation from which inputs and outputs can be selected from various parameters, field functions, reports, and monitors. The inputs are parametrized with a baseline value as well as an uncertainty distribution.

4.4.2 KACEGEN

KACEGEN (Kairos ACE GENERator) is a wrapper tool developed to drive NJOY21 to produce ACE-format nuclear data libraries for use in Serpent 2 and/or MCNP6.2. NJOY21 (Reference 43) is a nuclear data processing tool capable of producing both pointwise and multi-group cross section data from the U.S. Evaluated Nuclear Data Files (ENDF) format. KACEGEN can generate ACE libraries from any ENDF-6 format library. Kairos Power has generated several libraries, including JEFF 3.3, ENDF-B-VII.1, and ENDF-B-VIII.0. Both neutron cross-sections and thermal-scattering libraries are produced for each isotope available in the library, and thermal-scattering libraries can be discrete or continuous in energy (continuous only for ENDF-B-VIII). [[

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4.4.3 KPACS

KPACS is an internally developed core design and analysis wrapper code that simulates the pseudo steady-state evolution of the KP-FHR core by loosely coupling Serpent 2 and discrete element modeling (DEM) in STAR-CCM+. KPACS is specifically designed to operate in two modes.

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4.4.3.1 Zoner

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4.4.4 KPATH

KPATH is the Kairos Power-developed data transfer interface that connects STAR-CCM+ to Serpent 2.

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4.5 SCALE

The SCALE (Standardized Computer Analysis for Licensing Evaluation) code, developed by Oak Ridge National Laboratory (ORNL) (Reference 46), is a comprehensive suite of tools designed for nuclear reactor physics and radiation shielding analyses. The code includes a diverse range of modules, each tailored for specific applications, including lattice physics, burnup analysis, criticality safety, and radiation shielding. SCALE uses a vast library of nuclear data and cross-section data sets, providing accurate representations for various materials and isotopes. The code is widely used in the nuclear industry for reactor design, safety evaluations, and licensing assessments. SCALE also includes specialized modules such as TSUNAMI for sensitivity and uncertainty analysis, allowing users to assess the impact of input parameter uncertainties on FOMs. The Sampler module further enhances the capabilities of SCALE by providing a framework for generating covariance data and creating perturbed cross-section libraries for uncertainty quantification studies. The Sampler module also allows perturbation of input parameters, thereby making it a suitable framework for systems under development (i.e., KP-FHR) to investigate large parameter space. When as-built data becomes available, the module can also confirm or update calculations. Further discussion on uncertainty analysis framework and utilization of SCALE for this methodology is described in Section 5.2.

4.6 Software Quality

The software and computer codes used in the methodology described in this topical report are maintained under the Kairos Power software quality assurance program. The activities managed under the program include change management of the software and computer codes to evaluate impacts of updates and improvements along with error reporting and corrective action.

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Figure 4-1 High Level Process Flow Diagram of the Core Design and Analysis Methods

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Figure 4-2 **KPACS Data Flow Diagram**

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Figure 4-3 Example of Zoner Generated Spectral Zones Used for the KP-FHR Core

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Figure 4-4 KPATH Framework

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5 VALIDATION, VERIFICATION, AND UNCERTAINTY ANALYSIS

This section discusses the validation, verification, and uncertainty quantification of the discrete element methodology (Section 5.1), neutronics methods (Section 5.2), and thermal hydraulic methods (Section 5.3). The verification of the neutronic methods described in this section is an inherent part of the implementation of software quality assurance (Section 4.6). The use of nuclear reliability factors (NRFs) for neutronics only applies to those FOMs used as input into safety analysis as summarized in Section 3.6. This section also includes a discussion of the method for updating neutronic uncertainties (Section 5.4) as additional operational measurements are obtained.

The overall approach for validation of the discrete element modeling, neutronics, and thermal hydraulics methods is summarized by four strategies within the methodology: 1) use state-of-the-art high fidelity computer codes, 2) benchmark these codes against other high-fidelity codes, 3) apply conservative uncertainties, and 4) confirm the validation during the startup of the test reactor. These methods are detailed in the following sections.

5.1 Discrete Element Method (DEM)

The rigor of DEM validation for core physics applications [[
]] is commensurate with the impact on neutronics parameters. [[

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DEM modeling activities include sensitivity analyses, validation, and calibration for the expected steady-state ranges of KP-FHR operations. [[

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5.2 Neutronics

Applications of Serpent 2 for KP-FHRs are generally placed into the following categories:

- Safety analysis inputs
- Spatially dependent distributions (i.e., neutron/gamma flux and spectrum and associated FOMs)
- Burnup and decay
- Fuel loading and criticality predictions

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5.2.1 Serpent 2 Code-to-Code Benchmarking

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5.2.1.1 Simplified KP-FHR model

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5.2.1.2 Serpent 2 Code-to-Code Comparison [[

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5.2.1.3 Criticality

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5.2.1.4 Reactivity Coefficients

The isothermal temperature coefficient (ITC) is the derivative of reactivity with respect to changes in system temperature. [[

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The coolant void coefficient (CVC) is the derivative of reactivity with respect to decrease in coolant void density. [[

]] The coolant temperature coefficient (CTC) is the derivative of reactivity with respect to changes in Flibe coolant temperature. [[

]] The fuel temperature coefficient (FTC) is the derivative of reactivity with respect to change in TRISO kernel temperature. [[

]] The moderator temperature coefficient (MTC) is the derivative of reactivity with respect to changes in graphite pebble/TRISO material in the core. [[

]] The reflector temperature coefficient (RTC) is the derivative of reactivity with respect to changes in reflector graphite temperatures. [[

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5.2.1.5 Fission Reaction Rates

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5.2.1.6 Power Distribution

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5.2.1.7 Rod Worth

Reactivity control system (RCS) bank worth is calculated as the difference in multiplication factor from the reference case relative to a change in RCS rod position. [[

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The integral rod worth is calculated as the delta in multiplication factor with respect to the RCS bank rod worth position from the ARO position. The differential bank worth is the derivative of reactivity with respect to RCS bank position. Integral and differential RCS bank worth curves are shown in Figure 5-6 and Figure 5-7 respectively [[

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5.2.2.1 Criticality

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5.2.2.2 Reactivity Coefficients

The reactivity coefficient calculations are performed using the same approach described in Section 5.2.1.4. [[

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5.2.2.3 Power Distribution

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5.2.2.4 Rod Worth

RCS bank worth, integral rod worth, and differential bank worth are calculated using the same methods described in 5.2.1.7. [[

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5.2.2.5 Burnup

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5.2.3 Uncertainty Quantification

The sources of uncertainties considered both include the effect of design input uncertainties and numerical uncertainty, [[]]. To determine overall quantified uncertainty (QU) bounds [[]], impacts from independent sources are combined using the root-square method, as shown in Equation 40:

$$QU = \sqrt{\sigma_a^2 + \sigma_b^2 + \dots} \tag{40}$$

[[

]] The methods used for uncertainty quantification (UQ) use certainty tools based on [[]], which is described in Sections 4.5 and 5.2.3.1. A demonstration of the capabilities of the UQ tool are provided in Appendix A.

5.2.3.1 SCALE/Sampler Certainty Tool

Physics-based methodologies used for quantifying uncertainty propagate known uncertainties in input parameters [[

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5.2.3.2 Input Uncertainty

Input uncertainties are quantifiable uncertainties that include design input and operational uncertainties, manufacturing uncertainties and tolerances, nuclear data, material impurities, thermophysical properties, and other statistical variations due to modeling assumptions. [[

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5.2.3.3 Numerical Uncertainties

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5.2.3.4 Other Sources of Uncertainty

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5.2.4 Bias

Application of bias correction factors is intended to achieve closer alignment between calculational methods and measurements. Accounting for bias is a necessary step to ensure that the application of calculation predictions is conservative.

5.2.4.1 Comparison to Experimental Benchmarks

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The bias correction factors represent reasonably conservative approximations using the available data from benchmark. [[

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5.2.4.2 Applying Modeling Bias

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5.2.5 Margin and Nuclear Reliability Factors

The applications of the FOMs for input into safety analysis account for known sources of bias and quantifiable uncertainties. Margin is applied to FOMs in the form of Nuclear Reliability Factors (NRFs).
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5.3 Thermal-Hydraulics

This section describes the core thermal hydraulics model validation, verification, and an assessment of uncertainties [[

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5.3.1 Model Validation

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Section 3.5 provides additional information for the phenomena considered in these models and provides a background for the use of [[

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5.3.2 Uncertainties Quantification

The methodology includes the assessment and quantification of model uncertainties for steady-state core materials [[

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5.3.3 Model Validation [[

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5.3.4 Numerical Error and Solution Verification

Solution verification follows CFD best practices (References 28 and 29) and focuses on the assessment and minimization of the numerical discretization error δ_{NUM} and user effects. CFD computational mesh refinements assessments of numerical discretization error, and convergence of steady-state global and local mass, momentum, and energy are performed to minimize such error and quantify its impact.

5.3.5 Input Uncertainties

Model inputs sensitivity analyses are classes of simulations performed for every core thermal-hydraulics application [[

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5.3.6 Modeling Biases and Confidence Level Factors

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5.3.7 KP-FHR Testing

KP-FHR test reactor will monitor temperature measurements of reflector graphite, core outlet, reactor vessel inlet and outlet. Measurements will confirm the validity of the proposed methodology uncertainty assessment by ensuring temperatures are within the predicted bounding values.

5.3.8 Applicability of Porous Media Correlations

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5.4 Methodology for Updating NRFs With Operational Data

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5.5 Validation, Verification and Uncertainty Analysis Summary

The preceding sections have described the methodology for verification, validation, and uncertainty quantification for the DEM, neutronics, and thermal hydraulics methods for analysis of a KP-FHR.

The calibration methodology for DEM is performed by [[

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A method for updating the nuclear reliability factors is described in section 5.4, which relies on the use of measurement data from KP-FHR reactor operation. [[

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Table 5-1 Parameters for Code-to-Code Benchmark [[]]

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Table 5-2 Code-to-Code Comparison Model Naming Convention

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Table 5-3 Summary of Methodology for Code-to-Code Benchmark Parameters [[
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Table 5-4 Comparison of Multiplication Factor [[]]

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Table 5-5 Comparison of Calculated ITCs [[]]

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Table 5-6 Comparison of Calculated CVCs [[]]

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Table 5-7 Comparison of Calculated CTCs [[]]

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Table 5-8 Comparison of Calculated FTCs [[]]

[[]]

Table 5-9 Comparison of Calculated MTCs [[]]

[[]]

Table 5-10 Comparison of Calculated RTCs [[]]

[[]]

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Table 5-14 **ESM Detector Calculations for Reflector Regions (*subcritical case 1*)**

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Table 5-15 **ESM Detector Calculations for Reflector Regions (*subcritical case 2*)**

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Table 5-16 Comparison of Multiplication Factor [[]]

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Table 5-17 Comparison of Calculated ITCs

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Table 5-18 Comparison of Calculated CVCs

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Table 5-19 Comparison of Calculated CTCs

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Table 5-20 Comparison of Calculated FTCs

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Table 5-21 Comparison of Calculated MTCs

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Table 5-22 Comparison of Calculated RTCs

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Table 5-23 RCS Total Bank Worth Calculations

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Table 5-24 Summary of [[
Code Burnup Comparison

]] Components and Nuclear Data for Code-to-

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Table 5-25 Benchmarked Experiments Used for Bias Estimation

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Table 5-26 Bias Corrections

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Table 5-27 Sources of Unquantified Uncertainty

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Figure 5-1 COF Calibration Phase 1 Sensitivity Analysis

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Figure 5-2 COF Calibration Phase 2 Fully Packed Core

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Figure 5-3 Simplification for sKPH Model

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Figure 5-4 1-D Axial Fission Rate Distribution Comparison [[]]
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Figure 5-5 1-D Radial Fission Rate Distribution Profile Comparison [[

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Figure 5-6 Integral Bank Worth Comparison [[
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Figure 5-7 Differential Bank Worth Comparison [[
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Figure 5-9 Differential RCS Bank Worth [[]] Results Comparison
 [[

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Figure 5-10 Burnup Comparison [[]] for Single Fuel
Pebble Depletion in Pebble Zone R0Z0
 [[]]

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Figure 5-11 Burnup Comparison [[]] for Single Fuel
Pebble Depletion in Pebble Zone R2Z1
 [[]]

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Figure 5-12 Nuclear Uncertainty Quantification

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Figure 5-13 Geometrical Features of the High-Fidelity Packed Bed Model

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Figure 5-14 Porous Media Validation Framework

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6 APPLICATIONS

This section provides an overall description of the calculational methodology that are used within the core design codes and models described in Sections 3 and 4 of this topical report. This section also provides a description of the core physics parameters, which are used in downstream applications and as inputs into safety analysis, source term, and nuclear design.

6.1 Input into Safety Analysis

For neutronic parameters, enough cycles and histories per cycle must be run to, at a minimum, guarantee the convergence of the fission source distribution (i.e., Shannon entropy). However, an increased number of runs are used depending on the required rigor of the FOM for downstream applications. The Monte Carlo (MC) error is expected to be small relative to the uncertainty in FOMs calculated using methodologies described in Section 5.

6.1.1 Reactivity Coefficients

The calculations for reactivity coefficients are performed using Equation 52:

$$\alpha_x = \frac{\rho_{\Delta x} - \rho_{ref}}{\Delta x} = \frac{1}{\Delta x} \left(\frac{1}{k_{ref}} - \frac{1}{k_{\Delta x}} \right) \quad (52)$$

where α_x is the reactivity coefficient with respect to quantity x , Δx is the change in quantity x with respect to a reference condition (positive or negative), k_{ref} is the neutron multiplication factor of the core calculated from Serpent 2 at reference conditions, and $k_{\Delta x}$ is the neutron multiplication factor of the core calculated by Serpent 2 after x was changed by Δx . The KP-FHR reactivity feedback includes fuel temperature, coolant temperature, coolant void, moderator temperature, and reflector temperature as well as the isothermal temperature coefficient.

The fuel temperature coefficient (FTC), or Doppler coefficient, is calculated by changing the fuel kernel temperature. The coolant temperature coefficient (CTC) is calculated by changing the Flibe temperature and associated density. The Flibe temperature only changes in the pebble bed region, downcomer, core barrel gap, defueling region, and upper and lower plenum when there is a change to the coolant temperature coefficient. The coolant void coefficient (CVC) is calculated by changing the Flibe density in the same Flibe components as in the CTC calculation. Moderator temperature coefficient (MTC) is calculated by changing the temperature of all non-fuel material within the pebbles of the core (i.e., buffer, IPyC, SiC, OPyC, pebble fuel matrix, pebble core, pebble shell, and the graphite pebbles). The reflector temperature coefficient (RTC) is calculated by changing the entire graphite reflector temperature (i.e., top, bottom, and side reflector) and does not account for change in temperature dependent Flibe density. The isothermal temperature coefficient (ITC) is calculated by homogeneously varying the temperatures of the fuel, coolant, moderator, and reflector.

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6.1.2 Rod Worth

The reactivity shutdown system (RSS) design provides adequate reactivity worth to compensate for the following:

- Positive reactivity inserted from the decay of full power xenon
- Change in temperature from full-power conditions to safe shutdown temperature (i.e., power defect)
- Maximum operational excess reactivity
- Highest worth stuck element (i.e., fully withdrawn)
- Uncertainties, [[

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The maximum operational excess reactivity (which is controlled by the total fuel loading) is the additional reactivity available to the reactor and is used to change power levels or manage other transients such as a change in the mass flow rate, inlet coolant temperature, or depletion of absorber in the Flibe or reflector. The value of maximum operational excess reactivity is set for all core states. Element insertion limits will be defined, as needed.

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]] The control elements in the RCS are responsible for all planned, normal power maneuvers. The worth requirements depend on the KP-FHR design.

Rod worth is calculated using Equation 53 where $k_{eff,out}$ is the withdrawn position and $k_{eff,in}$ is the inserted position of interest. Differential control worth is calculated using Equation 54, where $k_{eff,i}$ is the neutron multiplication factor of the core for the control rod(s) position at step i , $k_{eff,i+1}$ is the neutron multiplication factor of the core for the control rod(s) position at step $i + 1$, z_i is the axial position of the control rod(s) at step i , and z_{i+1} is the axial position of the control rod(s) at step $i + 1$.

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6.1.3 Neutron Kinetics Parameters

In addition to reactivity coefficients, kinetics parameters such as delayed neutron fraction and their associated decay constant(s), neutron mean generation time, and neutron mean lifetime are also

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calculated and provided as input into safety analysis. The kinetics parameters are calculated using the iterated fission probability method (Reference 59) of Serpent 2.

Delayed photoneutrons, from Be (γ, n) reaction in Flibe, are also assessed to understand their impact on the effective delayed neutron fraction and delayed neutron group structure (Reference 60). This impact from delayed photoneutrons is smaller than other reactors (i.e., heavy water reactors) that have been impacted from this source of delayed neutrons.

6.1.3.1 Application of Bias

Kinetics parameters are obtained from Serpent 2 and used in the point kinetics model for safety analysis. Serpent 2, similar to MCNP6.2, uses adjoint weighted method to calculate these parameters, including delayed neutron fraction (β_{eff}) and prompt neutron lifetime (l_p). The results from Serpent 2 are verified using calculations of β_{eff} and l_p through the prompt method and the $1/v$ insertion method respectively (Reference 61). For verifying the prompt method calculation, β_{eff} is calculated using Equation 55:

$$\beta_{eff} = 1 - k_p * k^{-1} \quad (55)$$

where k_p is the prompt neutron multiplication factor and k is the multiplication factor for all neutrons in the system. The $1/v$ insertion method calculates the prompt neutron lifetime by calculating the perturbed prompt neutron lifetime l'_p for systems with trace amounts of $1/v$ absorber, as shown in Equation 56:

$$l'_p = \frac{1}{N_{B-10}\sigma_{a0}v_0} \frac{k_{ref} - k_{pert}}{k_{pert}} \quad (56)$$

where k_{ref} is the eigenvalue of the original system, k_{pert} is the eigenvalue of the system perturbed with trace amounts of B-10, N_{B-10} is the boron-10 number density in atoms/(barn-cm), v_0 is 220,000 cm/s, and σ_{a0} is 3837 barns at 220,000 cm/s. The prompt neutron lifetime of the original system is calculated as the limit of l'_p as the concentration of boron approaches zero. Kinetics parameters are dependent on the response time (e.g., at startup core and equilibrium core compositions), as well as the isotopic composition of the fission products when an equilibrium core is reached.

6.1.4 Power Distribution

Core methodology provides power distributions inside the vessel region. Table 3-5 summarizes the powers that are provided by the methodology inside the vessel. Table 6-1 describes the format, and Tables 6-2, 6-3, and 6-4 provide definitions of power peaking factors. [[

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6.1.5 Depletion Calculations

6.1.5.1 Fuel Cycle

KPACS, as described in Section 4.4.3, is used to calculate different core composition states (i.e., equilibrium and transition core). To sustain the desired level of under-moderation while maintaining acceptable excess reactivity, pebble bed reactors start with an initial composition that is a combination of lower enriched fuel (down to depleted uranium) and higher enriched fuel. The transition core is the state between initial core composition and final equilibrium core composition and starts by first removing lower enriched fuel and replacing by the higher enriched fuel. KPACS analyzes core composition at various time steps. The spectrum in the core evolves as the core composition progresses towards equilibrium. Operation within the boundaries set by excess reactivity (i.e., accounting for rod insertion limits, if applicable) ensures that the operational and safety-related parameters of transition core states are bounded by the end states (i.e., low-power and equilibrium cores).

6.1.5.2 General Depletion

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6.1.6 Spectrum Averaged Cross Sections

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6.2 Inputs into Nuclear Design

The tools used in Section 4 are used to provide input parameters in the areas identified in this section.

6.2.1 Sources of Reactivity

The evolution of Flibe and the structural graphite throughout the operational lifetime of a KP-FHR reactor has an impact on core reactivity, reactivity coefficients, and the neutron spectrum, which are used in the safety analysis. The presence of Li-6 and other impurities present in fresh Flibe and graphite at startup, as well as the addition of corrosion products, are considered in core modeling. Serpent 2 allows material definitions to be inserted and burnup analysis to be performed for relevant materials.

6.2.2 Flux Distribution

Neutron and gamma flux and associated spectrums are a direct output of Serpent 2 MC radiation transport simulations that use appropriate tallies. Distribution of neutron and gamma flux within the reactor vessel has secondary applications, such as power distribution calculations, cross section averaging, fluence to vessel internals, displacement per atom (DPA) calculations, and other similar applications. This capability is a direct derivative of defined tallies in Serpent 2 with appropriate energy structure.

6.2.3 Radiation Fluence on the Vessel and Internals and Radiation Damage

Serpent 2 is used to calculate the fast neutron fluence on the vessel and the internals, including but not limited to core barrel, pebble insertion line, graphite reflector, RCS, and RSS. Serpent 2 is also used to calculate helium (alpha) generation in the vessel and other metallic components within the vessel.

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6.2.4 Power Distribution

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6.2.5 Nuclear Stability

KP-FHRs are characterized by their high-power density. The fundamental design features of a KP-FHR (described in Section 2.1) result in a large neutron diffusion length (L) and neutron migration area (M),

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which are calculated using Serpent 2. [[

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6.3 Thermal Hydraulics

The two major applications of the core thermal hydraulics methodology are 1) characterization of core and reflector material temperatures for cross sections and 2) qualification boundaries for fuel and reflector graphite temperatures.

6.3.1 Core Temperatures

The methodology provides steady-state, best estimate, nominal core material temperatures and provides spatial distributions and characterizations of uncertainties. Core materials include Flibe, graphite pebbles, fuel pebbles, and TRISO particles. [[

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6.3.2 Reflector Temperatures

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6.4 Core Composition

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6.5 Core Follow

The tools described in Section 4 are used to predict reactor physics parameters for KP-FHR reactors during operation, encompassing the continuous refueling and burnup of fuel and Flibe, as the core undergoes evolution. The methodology provided in this report is used to account for the operational

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history (i.e., power and inserted pebble histories) and provides the predicted core composition. At a minimum, the comparison of the rod position is compared against predictions.

6.6 Startup Physics Testing

The startup process of a KP-FHR consists of two phases: fuel loading and zero power testing. The fuel loading process is the process in which a combination of fuel and moderator pebbles are loaded incrementally into the core. The first two steps of this process rely on the critical mass predictions using Serpent 2 and associated uncertainties. The first two insertions of fissile content into the core are limited to less than 50% of the lower bound of the critical loading estimate. Until the core achieves criticality, the process will use the 1/M approach for each step's prediction. The process of performing zero-power testing includes control rod calibration, shutdown rod worth, and isothermal temperature coefficient measurements are performed and are compared against predictions. If, during startup physics testing, predicted values with added uncertainties do not align with measured data, testing will be suspended. The impact on the safety analysis due to potential changes in nuclear reliability factor applicability will be evaluated before testing is resumed.

Once all zero-power physics testing is successfully completed, power ascension begins. Measurements of important reactor physics parameters will be repeated at designated hold points during power ascension, including but not limited to shutdown rod worth and control rod worth.

6.7 Example Limits and Margins for Selected Applications

A graphical summary of the thermal-hydraulics and DEM methodologies interfaces described in this report is presented in Figure 6-2.

DEM methodology is used for:

- 1) Generating baseline neutronics model geometry (pebble centers and spectral zones)
- 2) Generating porosity values for TH inputs

Thermal Hydraulics methodology is used for:

- 1) Providing temperatures for materials cross section evaluation in neutronics model
- 2) Ensuring operations within TRISO and reflector graphite temperature qualification

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Also provided in the table are the FOMs that are inputs to the safety analysis. These FOMs are shutdown margin, kinetic parameters, shutdown and control element worth, power distribution, and reactivity coefficients. [[

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Table 6-1 Power Format and Description

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Table 6-2 Global Peaking Factor

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Table 6-3 Axial Peaking Factor

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Table 6-4 Radial Peaking Factor

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Table 6-5 Steady state Hermes predictions compared to selected reference FOM limits.

Figure of Merit	Reference Limit	Example Hermes	Margin	Measured
Fuel Qualification Parameters (AGR-2)				
SiC Temperature [°C]	[[]]	[[]]	[[]]	[[]]
Burnup [%FIMA] [[]]	[[]]	[[]]	[[]]	[[]]
Particle Power [mW]	[[]]	[[]]	[[]]	[[]]
[[]]				
Peak Temperature [°C] [[]]	[[]]	[[]]	[[]]	[[]]
Peak Fluence x10 ²¹ [n/cm ²]	[[]]	[[]]	[[]]	[[]]
Inputs to Safety Analysis				
Shutdown margin [pcm] [[]]	[[]]	[[]]	[[]]	[[]]
Kinetic parameters	[[]]			[[]]
Shutdown and control element worth	[[]]			[[]]
Power distribution	[[]]			[[]]
Reactivity coefficient [[]]	[[]]			[[]]

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Figure 6-1 Core Composition Uncertainty Analyses

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Figure 6-2 Thermal Hydraulic-DEM models interfaces with downstream applications

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7 SUMMARY

An analysis method consisting of Serpent 2, STAR-CCM+, and the calculational framework associated with the codes described in Section 4 support the neutronics and thermal hydraulic design, steady-state analysis, and licensing of KP-FHRs. The software quality assurance program at Kairos Power is an integral part of the development and maintenance of this analysis framework.

The neutronics portion of this methodology relies on Serpent 2. [[

]] Additionally, such predictions will be confirmed during the fuel loading process and subsequent zero-power testing of Hermes and future KP-FHRs.

The thermal-hydraulics portion of this methodology produces temperature distributions across the vessel, including pebble bed and reflector geometry for a defined boundary. [[

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Both code-qualifying approaches (described in Section 5.2 and 5.3 for neutronics and thermal hydraulics respectively), are applicable to different KP-FHRs, given that the proper range of applicability is applied to the benchmarking methodology.

7.1 Conclusions

The DEM methodology described in Section 3.3 of this topical report is acceptable for generating both a random packed pebble bed and TRISO distributions to use for a Serpent 2 baseline model geometry (Section 3.4) and for thermal hydraulic validation benchmarks. These benchmarks, described in Section 5.3, are also used to inform the thermal hydraulic uncertainty quantification described in Section 5.3.3. This methodology is also able to adequately predict pebble tracks and velocity profiles during pebble recirculation. Pebble tracks and velocity profiles are used to generate core spectral zone boundaries for the fuel cycle analysis tool, KPACS, using Zoner (Section 4.4.3.1).

The method for using bounding DEM sensitivity analyses to assess the effect of COF on DEM models and predict the impact on FOMs is adequate for use in a KP-FHR. The sensitivity range is informed by internal tribology testing representative of the KP-FHR core conditions. During reactor startup physics testing, the DEM-COF calibration with respect to total number of pebbles in a fully packed core is performed. The final calibrated COF is used as baseline input for the DEM contact force model.

The thermal hydraulics methodology (described in Section 3.5) that is used to predict core material temperature distribution for Flibe, graphite pebbles, fuel pebble layers, TRISO layers, and reflector temperature distribution for applications described in Section 6.3 is also an adequate methodology for a

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KP-FHR. Predicted temperature values are used to model reactor temperature feedback in Serpent 2 models.

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The conservative nuclear reliability factors (NRFs) for the neutronics FOMs are determined using quantifiable uncertainties, bias derived from similar systems (see Table 5-26) and discretionary conservatism (see Table 5-27).

Based on these conclusions, this topical report provides an acceptable methodology for core design and analysis subject to the limitations in Section 7.2.

7.2 Limitations

Modeling tools and methods described in this topical report are applicable to KP-FHR steady-state operation. The methodology limitations include the applicability of the correlations used in the thermal-hydraulics modeling of the core.

This core design and analysis methodology is subject to the following limitations:

1. The methodology described is applicable to a KP-FHR consistent with the design described in Section 1.1.
 2. The methodology is applicable to steady-state conditions.
 3. Application of the methodology for future KP-FHRs will use measured data including isothermal temperature reactivity coefficient and the shutdown worth from a KP-FHR test reactor, to update nuclear reliability factors.
 4. The confidence level factor and bias for thermal hydraulic figures of merit are limited to KP-FHR test reactors. Use in power reactors will be justified using applicable separate effects tests or measured data from a KP-FHR test reactor.
 5. If discretionary conservatisms provided in Table 5-27, are updated for use in future KP-FHRs beyond Hermes, new discretionary conservatisms must be justified by KP-FHR measurement data.
 6. Ranges of applicability of the [[]]
- are determined by the methodology provided in Section 5.3.8.
7. The pebble velocity needs to be a small fraction of the time constant of delayed neutron precursors.
 8. The software and computer codes used are maintained under the Kairos Power software quality assurance program.

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APPENDIX A. Example Calculation

Calculation results for an example Hermes design are described in this appendix to illustrate the application of the methodology described in this topical report.

A.1 Neutronics

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A.1.1 Inputs to Safety Analysis

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A.1.1.1 Reactivity Coefficients

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A.1.1.2 Shutdown Margin

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A.1.2 Demonstration of Uncertainty Analysis Tool

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A.1.2.1 Input Design Space

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A.1.2.2 Demonstration of Sensitivity Study

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A.1.2.3 Determination of KP-FHR Core Composition Space

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KP-FHR Core Design and Analysis Methodology			
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A.1.2.4 Uncertainty Quantification for Inputs to KP-FHR Inputs to Safety Analysis

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A.1.3 Determining Bounds of Conservatism for Inputs to Safety Analysis Parameters

A.1.3.1 Nuclear Data and RSS Modeling Bias for ITC and RSS Total Worth

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A.1.3.2 Carbon Cross-Section and Thermal Scattering Nuclear Data Bias

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A.1.3.3 RSS Insertion Modeling Bias

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A.1.3.4 Applying Nuclear Reliability Factors to ITC

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A.1.3.5 Applying Nuclear Reliability Factors to Shutdown Worth

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A.2 Thermal Hydraulics

The thermal hydraulic example calculations are provided below reflecting the applications and figures of merit described in Section 6 of the topical report.

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A.2.1 Core temperatures characterization

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A.2.2 Reflector temperatures qualification

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KP-FHR Core Design and Analysis Methodology			
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Table A.1-1 Reactivity Coefficients Calculated for Hermes λ -and Ω -core States

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KP-FHR Core Design and Analysis Methodology			
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Table A.1-2 Hermes λ -and Ω -Core Shutdown Margin

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KP-FHR Core Design and Analysis Methodology			
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Table A.1-3 Total RCS/RSS Bank Worth

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Non-Proprietary	Doc Number	Rev	Effective Date
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Table A.1-4 Hermes Kinetics Parameters for λ -and Ω -core States

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Table A.1-5 Group-wise Delayed Neutron Fraction for λ -and Ω -core States

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KP-FHR Core Design and Analysis Methodology			
Non-Proprietary	Doc Number	Rev	Effective Date
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Table A.1-6 Group-wise Delayed Time Constant for λ -and Ω -core States

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KP-FHR Core Design and Analysis Methodology			
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Table A.1-7 Material Temperature Inputs and Distributions

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KP-FHR Core Design and Analysis Methodology			
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Table A.1-8 Material Density Inputs and Distributions

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KP-FHR Core Design and Analysis Methodology			
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Table A.1-9 Material Impurities Input and Heavy Metal Loading Distributions

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KP-FHR Core Design and Analysis Methodology			
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Table A.1-10 Uncertainty Quantification of ITC for Hermes λ -and Ω -cores

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KP-FHR Core Design and Analysis Methodology			
Non-Proprietary	Doc Number	Rev	Effective Date
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Table A.1-11 Uncertainty Quantification of RSS bank total worth for Hermes λ -and Ω -cores

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KP-FHR Core Design and Analysis Methodology			
Non-Proprietary	Doc Number	Rev	Effective Date
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Table A.1-12 Determined ITC and RSS Total Worth Nuclear Data and Modeling Biases for Hermes λ - and Ω -Cores

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KP-FHR Core Design and Analysis Methodology			
Non-Proprietary	Doc Number	Rev	Effective Date
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Table A.1-13 Comparison of Calculated Isothermal Temperature Coefficient for Considered Carbon Cross-Section and Thermal Scattering

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KP-FHR Core Design and Analysis Methodology			
Non-Proprietary	Doc Number	Rev	Effective Date
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Table A.1-14 Comparison of Calculated RSS Total Worth for Considered Carbon Cross-Section and Thermal Scattering Libraries

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KP-FHR Core Design and Analysis Methodology			
Non-Proprietary	Doc Number	Rev	Effective Date
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Table A.1-15 Determination of ITC Bounds of Uncertainty Based on Determination of Nuclear Reliability Factor

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KP-FHR Core Design and Analysis Methodology			
Non-Proprietary	Doc Number	Rev	Effective Date
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Table A.1-16 Determination of Shutdown Worth Bounds of Uncertainty Based on Determination of Nuclear Reliability Factor

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KP-FHR Core Design and Analysis Methodology			
Non-Proprietary	Doc Number	Rev	Effective Date
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Table A.2-1 Summary of Core TH Model Biases and Uncertainties for the Zone (R1, Z3)

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KP-FHR Core Design and Analysis Methodology			
Non-Proprietary	Doc Number	Rev	Effective Date
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Table A.2-2 Summary of Integral TH Model Biases and Uncertainties for Peak Reflector Temperature

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Figure A.1-1 RCS Integral Worth Curves for λ -and Ω -cores

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KP-FHR Core Design and Analysis Methodology			
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Figure A.1-2 RCS Differential Worth Curves for λ -and Ω -cores

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KP-FHR Core Design and Analysis Methodology			
Non-Proprietary	Doc Number	Rev	Effective Date
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Figure A.1-3 RSS Integral Worth Curves for λ -and Ω -cores

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KP-FHR Core Design and Analysis Methodology			
Non-Proprietary	Doc Number	Rev	Effective Date
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Figure A.1-4 RSS Differential Worth Curves for λ -and Ω -cores

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KP-FHR Core Design and Analysis Methodology			
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Figure A.1-5 Axial Fission Power Profile for Hermes λ -and Ω -cores Normalized Over Total Power

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KP-FHR Core Design and Analysis Methodology			
Non-Proprietary	Doc Number	Rev	Effective Date
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Figure A.1-6 Radial Fission Power Profile for Hermes λ -and Ω -cores Normalized Over Average Power Density

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KP-FHR Core Design and Analysis Methodology			
Non-Proprietary	Doc Number	Rev	Effective Date
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Figure A.1-7 Axial Flux Profile for Hermes λ -and Ω -cores Normalized Over Average Flux

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KP-FHR Core Design and Analysis Methodology			
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Figure A.1-8 Radial Flux Profile for Hermes λ -and Ω -cores Normalized Over Average Flux

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KP-FHR Core Design and Analysis Methodology			
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Figure A.1-9 [[]]

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Figure A.1-10 Axial and Radial Power Density Peaking Factor Distributions

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KP-FHR Core Design and Analysis Methodology			
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Figure A.1-11 2D Power Peaking Factor Distribution

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Figure A.1-12 Sensitivity of Effective Multiplication Factor with Respect to Lithium-6 Concentration in Flibe, Kernel Boron and HML and Reflector Boron

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KP-FHR Core Design and Analysis Methodology			
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Figure A.1-13 Sensitivity of Effective Multiplication Factor with Respect to Pebble/TRISO Boron, Reflector Density, and Flibe Density and Temperature

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KP-FHR Core Design and Analysis Methodology			
Non-Proprietary	Doc Number	Rev	Effective Date
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Figure A.1-14 Ranges of CVC with Respect to Flibe Lithium-6 Enrichment for Various Hermes λ -core Moderator Pebble Ratios

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KP-FHR Core Design and Analysis Methodology			
Non-Proprietary	Doc Number	Rev	Effective Date
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Figure A.1-15 Multiplication Factor Versus Moderator Pebble Ratio for Various Flibe Lithium-6 Concentration (optimal moderation)

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Figure A.1-16 RSS Modeling Considered for Total Worth Bias Study

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KP-FHR Core Design and Analysis Methodology			
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Figure A.1-17 [[]]

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KP-FHR Core Design and Analysis Methodology			
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Figure A.1-18 Sensitivity of Effective Multiplication Factor with Respect to Flibe, TRISO-Kernel, and Reflector Temperatures

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KP-FHR Core Design and Analysis Methodology			
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Figure A.2-1 Spectral Zones Flibe and TRISO Kernel Temperature Outputs from Input Sensitivity Analyses

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KP-FHR Core Design and Analysis Methodology			
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Figure A.2-2 Axial Reflector Temperature Boundaries (top) and Maximum Axial Fluence Distribution (bottom)

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KP-FHR Core Design and Analysis Methodology			
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APPENDIX B. Verification and Validation

B.1 Discrete Elements Modeling

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B.2 Thermal-Hydraulics

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B.2.1 Porous Media Closures Applicability

B.2.1.1 Pressure Drop

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B.2.1.2 Pebble-Flibe Heat Transfer Coefficient

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KP-FHR Core Design and Analysis Methodology			
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B.2.2.1 [[]]

B.2.2.1.1 Bed Heat Transfer

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B.2.2.1.2 Reflector Heat Transfer

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KP-FHR Core Design and Analysis Methodology			
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KP-FHR Core Design and Analysis Methodology			
Non-Proprietary	Doc Number	Rev	Effective Date
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Table B.1-1 Pebble Bed Cylinder Configurations

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KP-FHR Core Design and Analysis Methodology			
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Table B.2-3 Flow Rates and Pebble Powers for Pebble Bed Case Cylinder 2

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KP-FHR Core Design and Analysis Methodology			
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Figure B.1-1 Total number of pebbles inside the core vs DEM COF

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KP-FHR Core Design and Analysis Methodology			
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Figure B.1-2 [[

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KP-FHR Core Design and Analysis Methodology			
Non-Proprietary	Doc Number	Rev	Effective Date
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Figure B.2-1 Flibe and pebble surface temperature axial profiles [[
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KP-FHR Core Design and Analysis Methodology			
Non-Proprietary	Doc Number	Rev	Effective Date
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Figure B.2-2 Flibe and reflector surface temperatures axial for Re = 250 and reflector heating

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KP-FHR Core Design and Analysis Methodology			
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APPENDIX C. Neutronics PIRT Results for the KP-FHR

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KP-FHR Core Design and Analysis Methodology			
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