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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

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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 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 **[[** [redacted] **]]** 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]]] The NRC staff notes that the [[[redacted]]] was developed for the high-temperature gas-cooled reactor (HTGR). [[[redacted]]] TR section 3.5.1.1, "Pressure Drop Correlation," identifies the [[[redacted]]] TR section 3.5.1.1 also identifies [[[redacted]]]

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

The [[[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]]] 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]]] model. At this interface, [[[redacted]]] The NRC staff considers this to be a reasonable assumption for normal steady-state conditions because the [[[redacted]]] TR section 3.1.3 also states that, if necessary, [[[redacted]]] Therefore, the NRC staff determined that the overall treatment of boundary conditions at the LTNE PM model and CFD [[[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 data[1] 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] **]]** 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 [[[redacted]]] 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+. [[[redacted]]] 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]

[redacted]]] TR section 5.2.3 presents SCALE/Sampler [[[redacted]]]. Quantified uncertainties (QUs) in FOMs are developed based on [[[redacted]]] TR section 5.2.4, “Bias,” describes the development of bias in the Serpent 2 predictions based on [[[redacted]]] The NRC staff would typically expect more applicable experiments to be used in a benchmark suite. However, the staff notes that the [[[redacted]]] experiments chosen for validation listed in TR table 5-25, “Benchmarked Experiments Used for Bias Estimation,” are reasonable because they represent the best currently available data and directly applicable data will be collected from future operation of the Hermes test reactors as described in TR section 5.4, “Methodology for Updating NRFs With Operational Data.” The bias terms shown in table 5-26, “Bias Corrections,” are conservative relative to the data. TR section 5.2.5, “Margin and Nuclear Reliability Factors,” describes the process by which [[[redacted]]] The acceptability of this aspect of the methodology is dependent on the accuracy of the calculated components [[[redacted]]]

[[[redacted]

- [redacted]
- [redacted]]]

The TR asserts all the comparisons are within two standard deviations. The NRC staff notes that a more mathematically rigorous code-to-code comparison would focus on the difference in BE values and minimizing additional numerical uncertainty by tightly converging MC statistics through running a sufficiently high number of particles in the simulation. One measure of MC convergence is to show that the associated uncertainty is acceptably low relative to the quantity of interest being calculated with the MC model. The [[[redacted]]] However, in practical terms, it is difficult to converge differences (i.e., temperature coefficients are calculated via differences in k_{eff} between a reference and a perturbed state) as opposed to primary quantities (e.g., a single k_{eff} estimation), especially when the magnitude of the effect is low like for the RTC. The NRC staff notes that the RTC is positive, but it will always be offset by the much more negative MTC and fuel temperature coefficient (FTC) such that high uncertainty in the calculated values is relatively unimportant. Also, for safety-related applications, the estimated MC convergence uncertainty is factored into the overall uncertainty accounted for in the method. Therefore, the NRC staff found that the comparisons presented in TR section 5.2.1 provide confidence in Serpent 2’s abilities to predict safety analysis-relevant parameters to an acceptable accuracy.

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] **]]** 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] **]]** Figure 5-13, “Geometrical Features of the High-Fidelity Packed Bed Model,” of the TR shows an example of **[[** [redacted] **]]** as described in TR section 5.3.2, “Uncertainty Quantification.”

TR section 5.3.8, “Applicability of Porous Media Correlations,” describes the **[[** [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] **]]** 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] **]]** are also measured during operation. The NRC staff notes that 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]]] 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.

REFERENCES

1. Safety Standards of the Nuclear Safety Standard Commission (KTA), “Reactor Core Design of High Temperature Gas-Cooled Reactors, Part 3: Loss of Pressure through Friction in Pebble Bed Crores,” Technical Report, KTA 3102.3, dated March 1981
2. K. Wakao, S. Kaguei and T. Funazkri, "Effect of Fluid Dispersion Coefficients on Particle-to-Fluid Heat Transfer Coefficients in Packed Beds," Chemical Engineering Science, Vol. 34, pp. 325-326, 1978
3. S. Yagi and N. Wakao, "Heat and Mass Transfer from Wall to Fluid in Packed Beds," A.I.Ch.E., Vol. 5, No. 1, pp. 79-85, 1959
4. Hassan, Y. and Kang, C. “Pressure Drop in a Pebble Bed Reactor Under High Reynolds Number,” Nuclear Technology, 2012
5. Yuan, H., et. al. “High-Fidelity CFD Simulation of Mixed-Convection in a Pebble Bed Test Reactor Core,” Nuclear Technology, 2025
6. Hu, G., O'Grady, D., Zou, L. and Hu, R., 2020, Development of a Reference Model for Molten-Salt Cooled Pebble-Bed Reactor Using SAM (No. ANL/NSE-20/31). Argonne National Lab
7. Novak, A.J., Schunert S., Carlsen R.W., Balestra P., Andrs D., Kelly J., Slaybaugh R.N., Martineau R.C., H.D., 2020, Pronghorn Theory Manual (No. INL/EXT-18-44453-Rev001), Idaho National Lab
8. NEA, Best Practice Guidelines for the Use of CFD in Nuclear Reactor Safety Applications –Revision, OECD Publishing, Paris, 2015
9. ASME Standard for Verification and Validation in Computational Fluid Dynamics and Heat Transfer V&V 20 - 2009(R2021)

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