Enclosure 8

KP-FHR Core Design and Analysis Methodology, KP-TR-017-NP (Non-proprietary)

KP-TR-017-NP

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KP-FHR Core Design and Analysis Methodology

Technical Report

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0	Initial Issuance	September 2021
1	Revision updates Section 5.3.1, Section A.2.3, Table A-4, Table A-5, Table A-10 and Figure A-7 to address feedback from NRC audit	September 2022

EXECUTIVE SUMMARY

Kairos Power is pursuing the design, licensing, and deployment of a Fluoride Salt Cooled, High Temperature (KP-FHR) test 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 and thermal hydraulic analysis of the KP-FHR.

The KP-FHR core design methodology is comprised 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 wrapper codes. The verification and validation (V&V) methodology for Serpent 2 and STAR-CCM+ codes is described. The methodology is informed by a Phenomena Identification and Ranking Table (PIRT) evaluation.

Serpent 2 and STAR-CCM+ and the associated wrapper codes are used to calculate core composition at various phases of operation and corresponding parameters such as core reactivity coefficients, control and shutdown element worth, shutdown margin, power distribution and thermal hydraulic parameters. The scope of this report applies to normal operation and postulated events. The methodology for using the codes to perform these calculations and the limitations on the use of this methodology are provided. In addition, a methodology for calculating the uncertainty in these calculations is provided. Sample neutronic and thermal hydraulic results for a KP-FHR are provided as an appendix.

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NOMENCLATURE

Abbreviations/Acronyms	
ACE	A Compact ENDF
AHTR	Advanced High-Temperature Reactor
AOO	Anticipated Operational Occurrence
CE	Continuous Energy
CFD	Computational Fluid Dynamics
CSAS	Criticality Safety Analysis Sequence
DEM	Discrete Elements Method
DHRS	Decay Heat Removal System
ENDF	Evaluated Nuclear Data Files
FHR	Fluoride-cooled High Temperature Reactor
FOM	Figure-of-Merit
HTGR	High Temperature Gas Reactor
IET	Integral Effect Test
IPyC	Inner Pyrolytic Carbon Layer
KP	Kairos Power
KPACS	Kairos Power Advanced Core Simulator
KPATH	Kairos Power Advanced Thermal Hydraulics
MG	Multi-group
NRC	Nuclear Regulatory Commission
OPyC	Outer Pyrolitic Carbon Layer
PDC	Principal Design Criteria
PIRT	Phenomena Identification and Ranking Table
QA	Quality Assurance
RSS	Reactivity Shutdown System
SARRDL	Specified Acceptable System Radiological Release Limit
SET	Separate Effect Test
SiC	Silicon Carbide
TH	Thermal Hydraulics
TRISO	Tri-structural Isotropic
TSL	Thermal Scattering Law
UA	Uncertainty Analysis
V&V	Verification and Validation

1 INTRODUCTION

Kairos Power is pursuing the design, licensing, and deployment of a Fluoride Salt Cooled, High Temperature Reactor (KP-FHR). This report is being submitted in support of a construction permit application being submitted in accordance with 10 CFR 50.34(a), "Preliminary Safety Analysis Report," for a non-power test reactor known as Hermes. This report describes the core design and analysis methodology for the KP-FHR test reactor at the beginning of life, startup, power ascension and at equilibrium conditions. This methodology is uses as an appropriate means to develop and analyze the core design for normal operation and downstream use in nuclear safety analysis for the Kairos Power test reactor.

1.1 FHR DESIGN FEATURES

The KP-FHR test reactor is a graphite moderated, randomly packed pebble-bed reactor with molten fluoride salt coolant operating at high temperature and near-atmospheric pressure (Reference 1). The fuel in the KP-FHR is based on the Tri-Structural Isotropic (TRISO) carbonaceous-matrix coated particle design. The fuel kernel and some of the coatings on the particle fuel provide retention of fission products. TRISO particles are dispersed within the graphite matrix of fuel pebble's fuel layer. The KP-FHR fuel pebbles are buoyant in reactor coolant under steady state and postulated events. The reactor coolant is a chemically stable molten fluoride salt mixture, $2LiF:BeF_2$ (Flibe) enriched in Li-7, which also provides retention of fission products that escape from any fuel defects. A pebble handling and storage system (PHSS) continuously inserts pebbles at the bottom of the reactor core and extracts them from the top of the reactor vessel during normal operations. Pebbles are examined for burnup and damage and are either returned to the vessel or directed to storage.

A primary coolant loop circulates the reactor coolant using pumps and transfers the heat to an intermediate coolant loop via a heat exchanger for direct rejection to the atmosphere. The design includes a decay heat removal system (DHRS) operating passively above a threshold power. The DHRS relies on natural circulation within the vessel to transfer heat from the core to the DHRS through thermal radiation and convection heat transfer from the outer vessel wall to the DHRS. A set inventory of water in the DHRS is passively boiled off over the duration of a postulated event in which the primary heat transfer system is unavailable.

Fission product control in the KP-FHR test reactor relies primarily on the multiple barriers within the TRISO fuel particles and fuel pebble to ensure that the dose at the site boundary as a consequence of postulated events meets regulatory limits. Additionally, the molten salt reactor coolant serves as a distinct secondary barrier providing retention of solid fission products that escape the fuel particle and fuel pebble barriers. This additional retention is a key feature of the enhanced safety and reduced source term in the KP-FHR.

Reactivity control in the KP-FHR test reactor is accomplished primarily by insertable control elements and shutdown elements. The shutdown elements directly insert into the packed pebble bed core and the control elements insert outside the pebble bed into the nearby side graphite reflector. For planned power maneuvers of the KP-FHR reactor, only the control elements are used.

1.2 REGULATORY BACKGROUND

1.2.1 10 CFR Requirements

Nuclear Regulatory Commission (NRC) regulations in 10 CFR 50.34(a)(4) and (b)(4) requires an analysis and evaluation of the design and performance of structures, systems, and components 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 structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents.

The methodology described in this report is used to analyze the fuel and core during normal operation and postulated events.

1.2.2 Principal Design Criteria

The principal design criteria that apply to a KP-FHR test reactor are contained in the "Principal Design Criteria for the Kairos Power Fluoride Salt-Cooled High Temperature Reactor Topical Report" (Reference 2). While these principal design criteria (PDC) do not apply directly to the methodology, the core and analysis methodology is used to perform the necessary analyses which demonstrate compliance of the design with the following PDC. The following PDC are relevant:

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.*

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.*

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.*

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.*

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 anticipated operational occurrence, accounting for a single malfunction of the reactivity control systems.*

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.

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.

The methods described in this topical report are used to calculate the power distributions which are an input to the fuel performance calculations that assure that specified acceptable system radionuclide release design limits (SARRDLs) will be met as described in PDC 10. Similarly, core design methods are used to calculate the reactivity coefficients to assure that the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity as described in PDC 11. The inherent characteristic of the KP-FHR test reactor (small core and long neutron diffusion length) ensure that power oscillations do not result in conditions exceeding SARRDLs as described in PDC 12. The KP-FHR uses a functional containment to ensure that radiological releases to the public are within required limits as described in PDC 16. The methods described in this report provide the input for the fuel performance calculation that assures that the barriers to radiological release from the fuel are not compromised. PDC 25 requires that the protection system be designed to ensure that the SARRDL is not exceeded for anticipated operational occurrences and the methods in this report are used to support that design. Shutdown margin calculations performed with the methodology in this topical report ensure that the requirements of PDC 26, Reactivity Control Systems are satisfied. Finally, PDC 28 requires that reactivity systems are designed such that the amount and rate of reactivity addition cannot result in damage to the reactor coolant boundary or to the core, its support structure, and other reactor vessel internals. The nuclear design methods described in this topical report are used to support the assessment that the KP-FHR meets PDC 28.

2 CORE PHYSICS AND DESIGN

The core design and analysis methodology aligns very closely with the physical behavior of a KP-FHR core. The following section describes the reactor core physics of the KP-FHR and will serve as reference for the description of the modeling tools and capabilities used in core design.

2.1 DESCRIPTION

The KP-FHR core contains thousands of randomly packed buoyant fuel pebbles that slowly ascend through the reactor core. Pebbles are continuously inserted at the bottom of the reactor and extracted from the top. The dynamics of the reactor core is characterized by the transition from a startup core to an equilibrium core over time. The fuel pebbles may contain natural uranium all the way up to 19.55 wt% U-235 to reduce effective enrichment and core reactivity in early startup core operations. Depending on the chosen startup and operational schemes, the core will also contain a fraction of graphite-only moderator pebbles to maintain the desired carbon to heavy metal atom ratio. Similar to the water to fuel volume ratio in light water reactors, the carbon to heavy metal atom ratio is used in FHRs to define the neutron moderation conditions (over-moderated or under-moderated) and the mixing of different pebble types facilitates maintaining the core in under-moderated conditions.

When defining the desired carbon to heavy metal 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) but also to ensure negative coolant temperature feedbacks.

An increase in 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 Flibe. The latter effect is a function of the carbon to heavy metal ratio; therefore, the combined reactivity feedback can be designed to be negative by controlling the carbon to heavy metal ratio. After some period of operation, Li-6 is consumed and its concentration is lower than in fresh Flibe. Nevertheless, lithium-6 in Flibe is also produced by (n,α) reactions on Be-9 leading eventually to an equilibrium concentration.. Salt impurities present in fresh Flibe are also parasitic absorbers in addition to the accumulation of other corrosion material, each of which have an impact on the coolant reactivity coefficients. The properties and specifications for the reactor coolant are described in "Reactor Coolant for the Kairos Power Fluoride Salt-Cooled, High Temperature Reactor" topical report (Reference 18).

The ability to control the mixture of pebble types in the core allows excess reactivity to be minimized during startup and operation. Core reactivity is also controlled by the movement of the control elements. Shutdown elements are also available for insertion for safe shutdown at all core states. The KP-FHR thermal energy transfer phenomena in the core are described in Figure 2-1. 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. Figure 2-1 shows these heat transfer modes and those outside the reactor core as well. Bypass flow, core barrel, downcomer, reactor vessel and decay heat removal heat transfer mechanisms are also highlighted in this figure.

A portion of the reactor coolant flow, referred to as the bypass flow fraction, does not flow through the core. The thermal energy balance within the reactor core determines the temperature distribution within the fuel, moderator and the coolant that flows through the core. Larger bypass flow fractions result in higher coolant temperatures in the core. The temperature distribution is important because temperature can influence core reactivity levels, burnup, and power shapes.

2.2 FUEL DESCRIPTION

2.2.1 TRISO Particles

The KP-FHR TRISO particles include a UCO kernel, a porous carbon buffer layer, and inner pyrolytic carbon (IPyC), silicon carbide (SiC), and outer pyrolytic carbon (OPyC) layers, as shown in Figure 2-2. TRISO particles are overcoated with a mixture of natural and synthetic graphite, and resin binder material. The KP-FHR fuel particle kernels are composed of UCO, a mixture of UO_2, UC, and UC_2. The overcoat thickness is specified to produce a nominal 37% particle packing fraction after isostatic pressing and heat treatment in the pebble annular fuel region resulting in ~16,000 particles per KP-FHR pebble.

2.2.2 KP-FHR Fuel Pebbles

The KP-FHR fuel pebble design is 40mm in diameter and has three regions with specific functions that complement the pebble-bed FHR design, which is shown in Figure 2-2. The inner-most region of the KP-FHR fuel pebble contains a low-density carbon matrix core. The function of this region is to make the pebble buoyant in the Flibe coolant. An annular fuel region shell is located on the surface of the inner carbon matrix core. This region is composed of a carbon matrix embedded with TRISO fuel particles. The fact that the fuel particles are closer to the pebble surface than in other designs (e.g., high temperature gas reactor) reduces the fuel temperatures relative to those designs. A fuel-free carbon matrix shell is located on the surface of the fuel region to protect the fuel region from mechanical damage during handling and operation.

2.3 REACTOR CORE DESIGN

An axial section of the Hermes reactor can be seen in Figure 2-3. In a typical KP-FHR reactor core, there are a few design features present that need to be captured in the analysis: the cylindrical pebble bed region, the upper and lower conic regions, the fueling region, the defueling chute, coolant inlet and outlet channels in the reflector, bypass and engineered channels in the reflector, and the reactivity control and shutdown system (RCSS).

The core is the region of the pebble bed that produces considerable fission power density, determined as the region from the top of the upper conic region of the core to the bottom of the lower conic region of the core. Core geometrical characteristics such as the conic regions and defueling chute are designed to support a more uniform burnup and fuel performance in the core, as the conic regions and relative diameters of defueling chute and cylindrical section impact pebble velocity profile and residence time. The function of the fueling region, located at the bottom of the reactor core, is to guide the pebbles coming from the insertion line(s) into the reactor core. The defueling chute, located at the top of the reactor core, is also designed to be a low power producing region where pebbles have adequate amount of time to allow for the decay of short-lived fission products. The decay heat generation is then low enough for the pebble handling system to operate within designed temperature limits to accept the extracted pebble.

Coolant inlet and outlet channels located in the bottom and top reflector, respectively, are designed to reduce pressure losses while achieving acceptable flow distribution and flow rates through the core. The block-type reflector design is characterized by the presence of radial and axial spoke gaps between blocks and at the interface with the vessel core barrel. This geometry causes a portion of the mass flow rate to bypass the core region. Engineered channels in the graphite blocks allow for the movement of reactivity control elements placed ex-core and any additional channels required to reduce temperature in the reflector.

The reactivity control and shutdown system consist of control elements that insert directly into the reflector (near the periphery of the core) and shutdown elements that directly insert into the pebble bed. The control elements are credited only for all planned power maneuvers of the KP-FHR reactor. To achieve short-term shutdown (i.e. not considering delayed impact from xenon), only the control elements are needed. To achieve safe shutdown conditions, the shutdown elements are used assuming the highest worth shutdown element is fully withdrawn (stuck). The design of the reactivity control and shutdown system must satisfy PDC 25 and PDC 26 (Reference 2).

2.4 OPERATIONAL REGIMES

There are four main periods of core operation in the life of the KP-FHR reactor with respect to criticality and composition: startup, power ascension, approach to equilibrium core, and equilibrium core. An illustration of these stages can be observed in Figure 2-4.

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While still subcritical, source range control element worth testing is performed by measuring changes in neutron multiplication from a start-up source. The distribution of flux is also monitored and assessed against predicted calculations during this stage. Once criticality is achieved and at zero power, isothermal reactivity coefficient testing is performed and compared against predicted calculations.

Once all zero-power physics testing is completed, the ascension to the power phase begins. The primary salt pump runs at reduced speed to provide forced circulation. As the power level increases from zero power, negative reactivity feedbacks arise from temperature increases, the buildup of xenon, and the depletion of fuel. To compensate for these effects, the reactivity control elements can be partially withdrawn. This balance of excess reactivity and extraction of heat from the core continues until full power is reached.

At full power (or the initial power plateau), the approach to equilibrium core begins. For the initial core composition, the radionuclide inventory is mostly fresh fuel, and burnup has not yet accumulated. To compensate for accumulated burnup, fresh fuel pebbles are added, and depleted pebbles removed, at a rate that maintains core reactivity. After some period of power operation, the isotopic concentration will be largely unchanged, and a stable rate of insertion and extraction of fuel will be reached (assuming constant power). At this point, the equilibrium core has been reached, which is designed to stay within the designed coolant reactivity coefficients, power per particle limits, and the desired excess reactivity.

2.5 PHENOMENA IDENTIFICATION AND RANKING TABLE

A PIRT evaluation was conducted for the KP-FHR core design. A full review of the existing Georgia Institute of Technology FHR neutronics PIRT (Reference 3), which uses the Advanced High-Temperature Reactor (AHTR) reactor design as the basis, was performed prior to beginning the KP-FHR PIRT. The description of Figures of Merit (FOMs) and knowledge level numbering used in the PIRT are as follows:

- FOM 1: Multiplication factor (1: **L**ow impact, 2: **M**edium impact, and 3: **H**igh impact)
- FOM 2: Power distribution (1: **L**ow impact, 2: **M**edium impact, and 3: **H**igh impact)
- Knowledge: Knowledge level (1: **L**ow, 2: **M**edium, and 3: **H**igh)

A summary of the KP-FHR PIRT results are provided in Appendix B - Neutronics PIRT for the KP-FHR.

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The predictive capabilities requirements for the KP-FHR core thermal hydraulics (TH) modeling follow the most relevant core TH PIRT phenomena.

3 CORE MODELING

3.1 MODELING

The KP-FHR core configuration is heterogeneous and non-stationary. The pebble bed continuously evolves from an early startup phase to a statistically steady burnup equilibrium condition. KP-FHR core physical characteristics such as core geometry, heterogeneity, and pebble bed motion require unique modeling approaches. The core design also requires different modeling approaches due to the lack of existing KP-FHR data.

3.2 MODELING PARADIGM

The methodology developed for core analysis and design aligns very closely with the physical behavior of the core. The KP-FHR core model paradigm includes discrete elements methods (DEM), neutronics and TH modules with several degrees of explicit coupling between them.

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3.3 DATA FLOW

The KP-FHR steady state and pseudo-steady state core design modeling workflow and data exchange consists of different degrees of coupling between DEM, neutronics and TH modules. Figure 3-1 presents a graphical summary of the data flow and processing of the core modeling paradigm.

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3.4 MODELING BOUNDARIES AND OUTPUT PARAMETERS

The domains of interest for modeling are first determined before performing core analysis for calculating quantities of interest for reactor safety and for input into downstream use in safety analysis and source term calculations. Domains of interest are both the geometric and material boundaries that are considered. These domains of interest are defined for each of the DEM, neutronics, and thermal-hydraulics calculations and are shown in Figure 3-2. A representative figure showing the axial section of the Hermes vessel (with simplified internals) along with a list of the geometric and material boundaries for each calculation is provided in Figure 3-2.

DEM considers the core shape, reflector material, the pebbles, and the coolant flow through the core when performing 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 fueling region.

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Thermal-hydraulics calculations are also three-dimensional calculations [[

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Using these boundaries for core analysis, the calculated output quantities of interest from this methodology include the following:

- Reactivity coefficients
    - Fuel temperature
    - Moderator temperature
    - Coolant temperature
    - Coolant void
    - Reflector temperature
- Control and shutdown element worth
    - Integral worth
    - Differential worth
- Power distribution
    - Peaking factor
    - Axial and radial power profile
- Kinetics parameters
    - Effective delayed neutron fraction
    - Effective neutron mean generation time

- Prompt neutron mean lifetime

For postulated event modeling of the KP-FHR, the following data is used: reactivity coefficients, kinetics parameters, shutdown margin, differential rod worths, and power shape (axial and radial). The tools used, the methodology, and example calculation at equilibrium for each of these quantities are provided in Section 4, Section 5, and Appendix A, respectively. This data will be provided as inputs to safety analysis at the following core compositional regimes (see Section 2.4): startup and the equilibrium core. This can also be done for other core states between startup and equilibrium, as needed. Conservative selection of the applied associated uncertainties is used for each of these quantities for the purposes of postulated event analysis (upper or lower bound), and each parameter's associated uncertainty analysis methodology is described in Section 6.

The fuel composition at equilibrium core is also used for source term analysis. Conservative selection of burnup uncertainties are applied for the purposes of source term analysis (whether upper or lower bound), and the burnup uncertainty analysis methodology is described in Section 6.4.

4 CORE DESIGN TOOLBOX

In order to develop the modeling methodology described in Section 3 , a series of software codes are used along with different code “wrappers” for coupling and data exchange (see Figure 3-1). The software used by Kairos Power for core design includes Serpent 2 for the neutronic module and STAR-CCM+ for DEM and TH modeling. KACEGEN, Kairos Power Advanced Core Simulator (KPACS), and Kairos Power Advanced Thermal Hydraulics (KPATH) are internally developed “wrapper codes” that have been developed to process and exchange data between software codes and libraries. The software discussed in this report is developed and maintained under the Kairos Power Quality Assurance (QA) program.

The verification, validation and uncertainty quantification methodology has been developed to reduce and control all the sources of error and uncertainty between STAR-CCM+ models used in core design and their FOMs predictive capabilities. The verification process consists of software and numerical solution verification activities. Software verification aims to ensures that the discretized model is an accurate representation of the continuous mathematical model, and that there are no user-defined code errors. 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 high order numerical results.

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4.1 CODES

4.1.1 STAR-CCM+

Description

STAR-CCM+ is used for DEM and TH modeling. The verification, validation and uncertainty quantification methodology has been developed to reduce and control all the sources of error and uncertainty between STAR-CCM+ models used in core design and their FOMs predictive capabilities. The V&V methods for DEM and TH are very similar. The TH V&V methodology focuses on the prediction of the core material temperatures (fuel, moderator and coolant) whereas the DEM methodology focuses on the pebble center locations and their residence time within the core. Because design to reduce bypass flow is performed independently due to the complexity of bypass flow paths, bypass flow is treated as a defined fraction of total coolant flow which is an input parameter to core design and analysis.

V&V Plan

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Table 4-2 summarizes the validation cases that will be used for STAR-CCM+.

4.1.2 Serpent 2

Description

Serpent 2 (Reference 4) is the main neutronics tool for reactor core design and output to safety analysis. Serpent 2 has been extensively used across academia and industry and has been validated against various benchmarks. It is used at Kairos for a variety of calculations, including multiplication factor, control element worths, reactivity coefficients, power distribution, kinetics parameters, nuclear heating, and burnup calculations.

The use of Serpent 2 provides high-fidelity simulation, which is important due to lack of experimental FHR operating experience. There are two key features that are available in Serpent 2: 1) the ability to explicitly capture the double-heterogeneity of the fuel pebble and TRISO particles, and 2) the implementation of Woodcock delta-tracking (Reference 4). The reduced computational burden with the implementation of delta-tracking also allows for full 3D core modeling.

V&V Plan

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4.2 WRAPPER CODES

The following wrapper codes are used to transfer information between the neutronic and thermal/hydraulic codes.

4.2.1 KACEGEN

Description

KACEGEN (Kairos ACE Generator) is an internal tool that NJOY21 uses to produce the ACE-format nuclear data libraries. NJOY21 (Reference 9) is a nuclear data processing tool capable of producing both pointwise and multigroup cross section data from the U.S. Evaluated Nuclear Data Files (ENDF) format. KACEGEN, as an example, currently has the capability to generate ACE libraries from any ENDF-6 library, 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. ACE data has been generated at the following temperatures, tailored to the temperature ranges in the KP-FHR design: 273.15, 300, 600, 700, 800, 900, 1000, 1100, 1200, 1500, 1800, and 2200 degrees K.

The high-level data flow of the KACEGEN can be seen in Figure 4-1. To start, library fields and paths for a particular library set are loaded, then the complete list of isotopes are run and written for neutron cross section generation. If thermal scattering is also being generated, the LEAPRs module of NJOY21 is run for either continuous and/or discrete thermal scattering law (TSLs).

Point-wise cross sections are compared between LANL-MCNP and OECD evaluated libraries and the ones evaluated by KACEGEN/NJOY.

4.2.2 KPACS

Description

KPACS is an internally developed fuel cycle analysis wrapper that loosely couples Serpent 2 and discrete element modeling (DEM) in STAR-CCM+ for pseudo steady-state analysis. The major underlying assumption in KPACS is shared with past codes such as VSOP (Reference 10) and PEBBED (Reference 11), in that the behavior of neutron spectrum and temperature affecting pebbles in specifically defined regions of the core (i.e., spectral zones) can be assumed constant due to slowly varying neutron flux and temperature. [[

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The difference between KPACS and other codes is that it relies on the highest fidelity tools available. Serpent 2 is used to perform the full-core transport and fuel depletion calculations, and the pebble motion and locations are informed by DEM in STAR-CCM+. KPACS can also be loosely coupled with KPATH , for update of core temperature distribution as needed throughout the KP-FHR operational life.

4.2.3 KPATH

Description

KPATH is the internally developed software infrastructure that couples STAR-CCM+ to Serpent 2. The computational fluid dynamic (CFD)-neutronic steady-state two-way explicit coupling, is such that it provides TH feedbacks for criticality calculations, power shape, and power peaking calculations.

Within the KPATH computational framework, STAR-CCM+ is utilized as a steady state solver for heat transfer and fluid flow in the form of a 3D porous media model. Only normal steady state operating conditions are considered. The coupling methodology that has been developed is capable of using KPATH during all core phases from startup to equilibrium core conditions; this means that KPATH can be used at different core phases in combination with KPACS.

The KPATH code wrapper manages the thermal power and materials temperatures exchange between Serpent 2 spectral zones and STAR-CCM+ porous region as shown in Figure 4-2. Convergence is reached when k_{eff} and core material temperatures differences with the previous iteration is within a specified tolerance.

5 CALCULATION METHODOLOGY

This section provides an overall description of the models used within the core design and analysis model.

5.1 DEM MODELING

The Discrete Element Method (DEM) is the methodology used to generate the reactor core geometry for the explicit pebble modeling in Serpent 2. DEM is utilized to simulate the granular flow by describing the motion of many interacting discrete solid pebbles. DEM modeling provides detailed resolution that other methods cannot achieve. [[

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STAR-CCM+ models DEM pebbles based on a soft-particle formulation in which particles are allowed to develop an overlap proportional to the contact force and can undergo large deformations without rupture. This overlap is not physically realistic but is used to aid in ease of computation. The calculated contact force is proportional to the overlap, as well as to the particle material and geometric properties. STAR-CCM+ DEM provides a large amount of data for every single pebble such as time histories, velocity, position, and forces. The data collected provide a statistical basis for neutronic calculations. DEM provides the location of the centers of the fuel pebbles necessary for criticality calculations. Burnup calculations need more information in addition to the location of the individual pebbles. DEM provides the pebble flow profile inside the reactor core and pebble residence time. The methodology to calculate residence time is based on recirculation of the pebbles in the core from the entry to exit point.

The FOMs used for the DEM V&V plan are:

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5.2 NEUTRONICS

5.2.1 Monte Carlo Convergence

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5.2.2 Fuel Cycle Analysis

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5.2.3 Reactivity Coefficients

The calculation for reactivity coefficients is done using Equation 5-1. Where α_x is the reactivity coefficient with respect to quantity *x,* Δx is the change in quantity *x* with respect to reference conditions (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 quantity *x* was changed by Δx.

$$\alpha_x = \frac{\rho_{\Delta x} - \rho_{ref}}{\Delta x} = \frac{1}{\Delta x}\left(\frac{1}{k_{ref}} - \frac{1}{k_{\Delta x}}\right) \qquad \text{Equation 5-1}$$

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5.2.4 Control Worth and Shutdown Margin

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Shutdown margin is also maintained at all core states. The control elements in the RCSS are responsible for all planned, normal power maneuvers. The worth requirements depend on the KP-FHR design of interest.

Control worth is calculated using Equation 5-2 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 5-3, where $k_{eff,i}$ is the neutron multiplication factor of the core for step i position of interest, $k_{eff,i+1}$ is the neutron multiplication factor of the core for step $i+1$ position of interest, z_i is the axial position of the control rod(s) for step i position of interest, and z_{i+1} is the axial position of the control rod(s) for step $i+1$ position of interest.

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5.2.5 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 calculated. As discussed in Section 3.4, kinetics parameters are used for modeling time-dependent behavior of the KP-FHR. The calculation of these kinetics parameters is calculated using the iterated fission probability method (Reference 14). The effective delayed neutron fraction is divided into six groups.

Delayed photoneutrons, from Be (γ, n) reaction in Flibe, will also be assessed to understand their impact on the effective delayed neutron fraction and delayed neutron group structure (Reference 15). This impact from delayed photoneutrons is smaller than other reactors that have been impacted from this particular source of delayed neutrons, such as from heavy water (D_2O) reactors.

5.2.6 Reactor Coolant Depletion

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5.2.7 Power Distribution

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5.2.8 Vessel Irradiation

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5.3 THERMAL-HYDRAULICS

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5.3.1 Porous Media Modeling

Packed beds are commonly used in chemical engineering systems because they have highly predictable and uniform flow distributions and transport. The randomly packed pebble bed in the KP-FHR core enables the use of low-order mathematical models to predict global flow distributions and heat transport. The TH model used in core design, adopts a two-equation porous media model to describe the macroscale behavior of the flow and energy transport within the reactor core region. The core porous region can be thermally coupled with other in-vessel solid regions, including the reflector structure, by the use of conjugate heat transfer modeling. The TH model resolved porous length scales characterize the liquid/solid phase mixture of liquid coolant and solid pebbles. The core macroscale porous model is derived by applying a volume averaging operator to the Navier-Stokes and two phase energy transport equations over a representative elementary volume. The solid and liquid phases are assumed to be in non-thermal equilibrium; this allows modeling two separate temperature fields for the coolant and pebbles respectively. The solid porous phase representing the pebbles uses the fission power density from Serpent 2 as an energy source term and provides the pebble average surface temperature distribution.

By removing information about resolved geometry, the volume averaging operator generates additional unknown terms in the momentum and energy equations that need a modeling mathematical closure correlation. As is conventional, experimental-based correlations are used to model the local momentum and heat transfer information that are lost during the volume averaging process. Figure 5-3 shows an example of local heat transfer phenomena that need a closure correlation.

Momentum closure model

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$$P_v v_s + P_i \, v_s |v_s| \quad \text{Equation 5-9}$$

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Energy closure models

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5.3.2 Core Material Temperatures

The pebbles and coolant temperature fields result from the porous medium volume averaged Navier-Stokes equations and energy equations are used as baseline for the core material temperatures evaluation. The core material temperatures that the TH model provides to the neutronic module are:

- Flibe temperature
- Graphite pebble temperature
- Pebble layers material temperatures
  - Low-density core
  - Fuel matrix layer temperature
  - Shell layer
- TRISO layers material temperatures
  - Outer PyC layer
  - SiC Layer
  - Inner SiC Layer
  - Buffer Layer
- Fuel Kernel temperature

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The spatial temperature distributions calculated for each material type that are fed back into the neutronics module are volume-averaged based on zones for the level of fidelity that is required for the calculated output parameter of interest.

The flexibility of the TH module implementation allows the thermal coupling with any other reactor vessel internals of relevant neutronic importance, as a consequence additional core material temperatures can be added to the list above for explicit coupling.

5.4 CORE BYPASS MODELING AND REFLECTOR TEMPERATURE

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6 UNCERTAINTY ANALYSIS AND NUCLEAR RELIABILITY FACTORS

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6.1 NUCLEAR DATA UA PROPAGATION METHOD

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6.2 MANUFACTURING INPUTS UA PROPAGATION METHOD

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6.3 KINETICS PARAMETERS CALCULATION UA METHOD

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6.4 BURNUP CALCULATION UA METHOD

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7 SUMMARY

7.1 CONCLUSION

This report documents the core design and analysis methodology which is used to perform nuclear design and thermal hydraulic calculations for the KP-FHR, including reactivity coefficients, shutdown margin, power distribution, and reactor core kinetics parameters. These methods apply to normal operation and postulated events for a KP-FHR test reactor.

The methodology for performing core design and analysis is based primarily on the Serpent 2 and STAR-CCM+ codes. These high fidelity analytical tools are used in a methodology specifically tailored to the unique features of the KP-FHR.

V&V of the Serpent 2 and STAR CCM+ codes is performed through comparisons with experimental results and to analyses from other codes. The uncertainty in the results from these codes is established based on industry experience and with a conservative bias due to the lack of operating experience with KP-FHRs. The conservative determination of uncertainties is confirmed using the SCALE code system.

The completion of the V&V of the codes and methodology will be submitted to the NRC as part of a future licensing applications that makes use of this methodology. In addition, the values of the uncertainties used in any applications will be documented as part of the safety analysis documents associated with the application.

7.2 LIMITATIONS

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

1. The pebble velocity needs to be a small fraction of the time constant of delayed neutron precursors.
2. Range of coolant velocity is applicable to the range of the available heat transfer correlations.

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Table 4-1. Serpent 2 Requirements and Planned Code-to-code Benchmark Validation

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Table 4-2. STAR-CCM+ Validation Cases

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Table 4-2. STAR-CCM+ Validation Cases

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Table 6-1. Scope of Uncertainty Analysis for Core Safety Parameters

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Figure 2-1. Thermal Energy Transfer Phenomena in KP-FHR

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

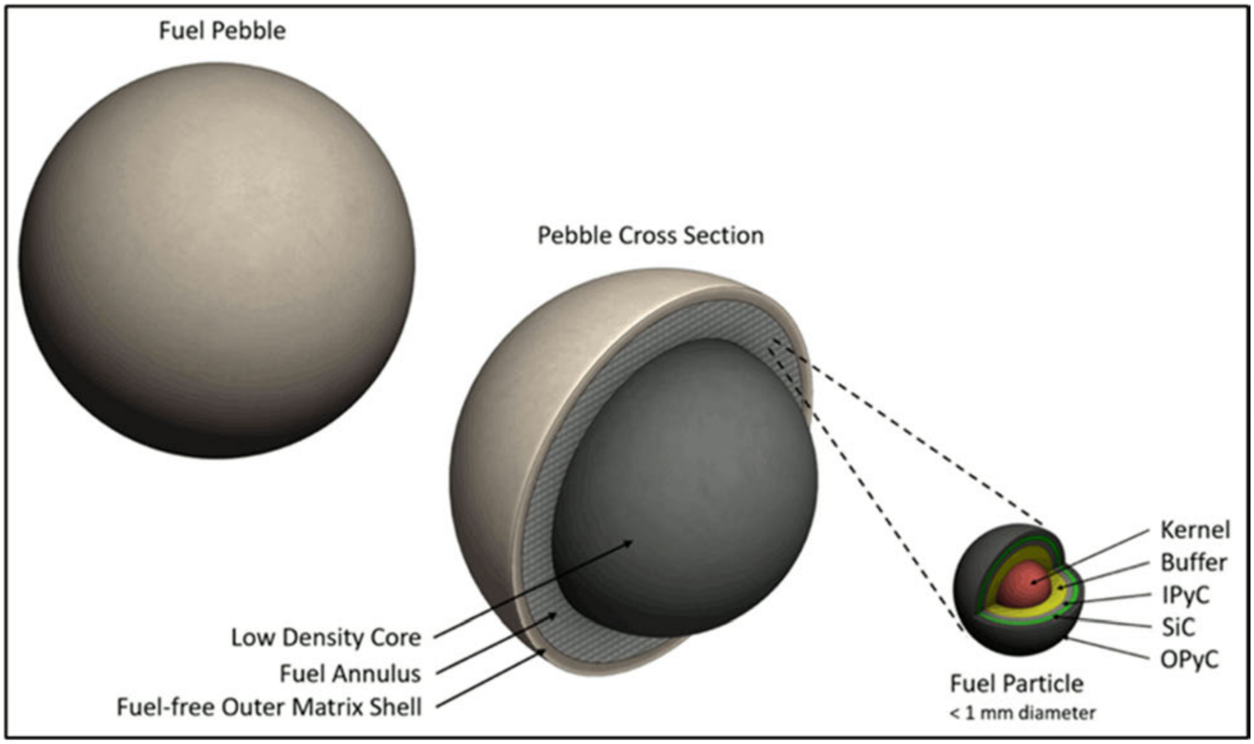

Figure 2-3. Axial Section of the Hermes Core

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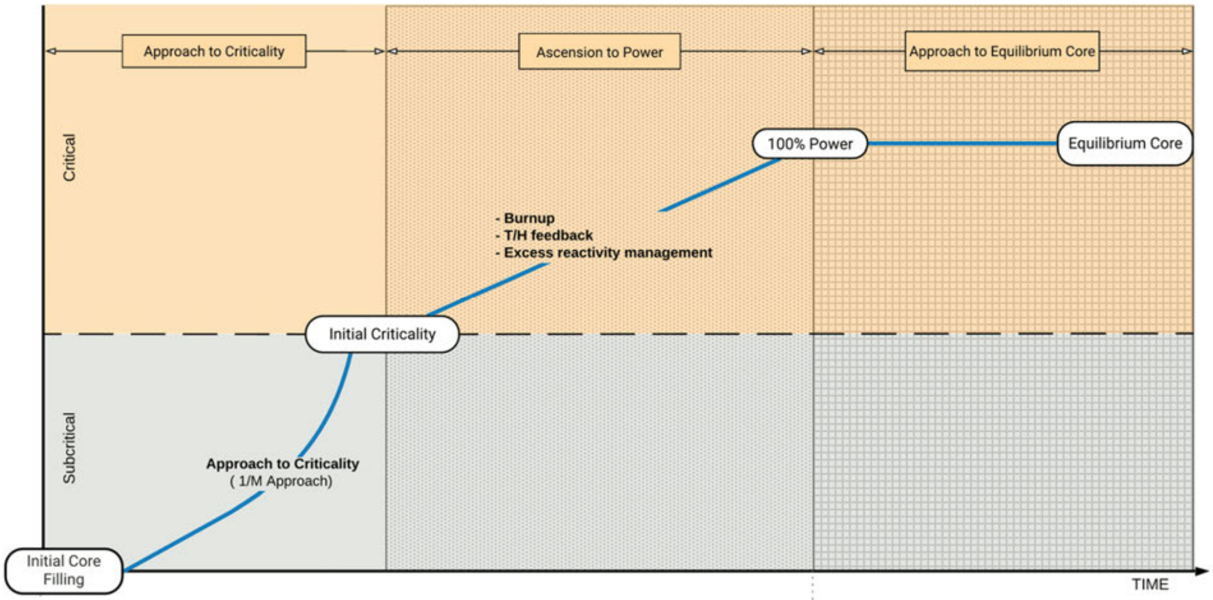
Figure 2-4. Reactor Core Operational Regimes of the KP-FHR



Note : The x- and y-axis are notional and are not to scale. Below the critical level on the y-axis represents subcriticality, and above the critical level on the y-axis represents power level, but each are notional.

Figure 3-1. Core Design Modules

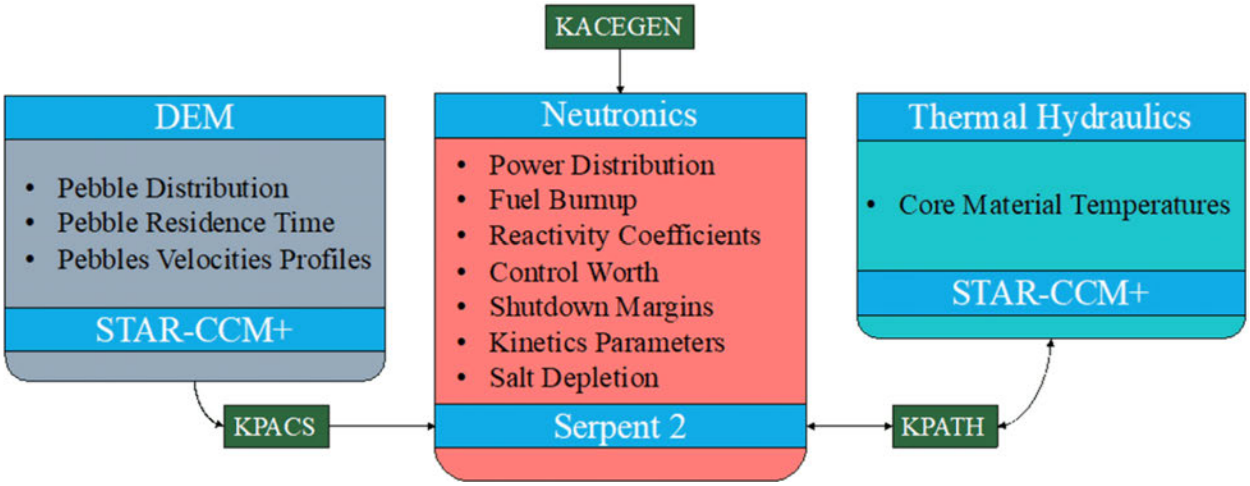


Figure 3-2. Core Analysis Boundaries

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Figure 4-1. High-level Data Flow of KACEGEN

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Figure 4-2. KPATH Framework

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Figure 5-1. Example Illustration of Algorithm for Pebble Circulation for the Core

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Figure 5-2. Spectral Zones Used for the Hermes Core

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Figure 5-3. Local Heat Transfer Phenomena in Pebble Bed Reactor Configuration

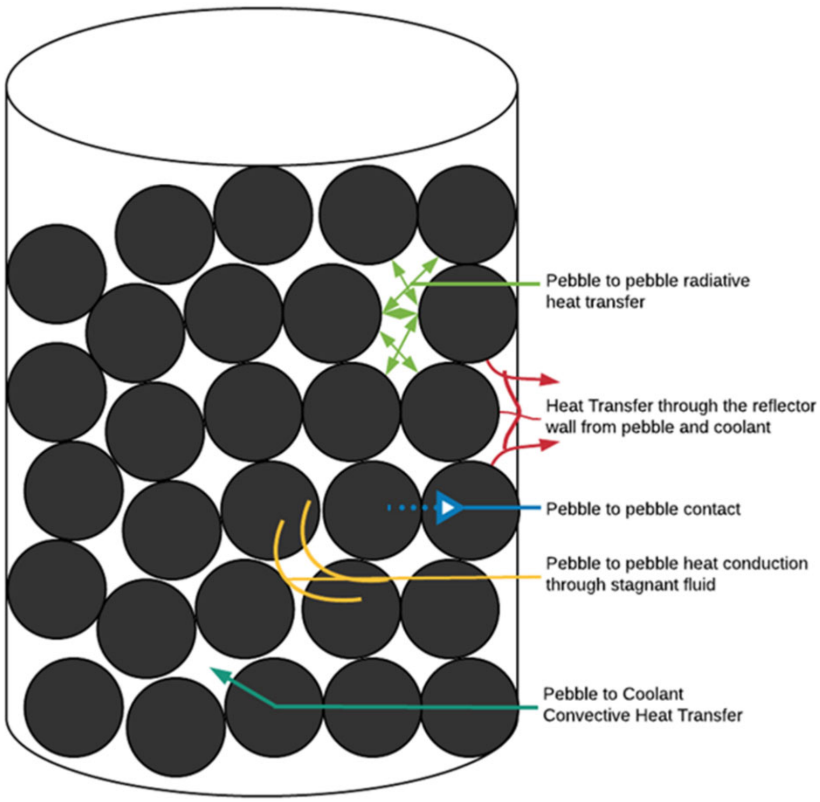

Figure 5-4. Pebble and TRISO Layers Temperature

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Figure 6-1. SCALE workflow for an Example Demonstration Involving Perturbed Parameters (in yellow)

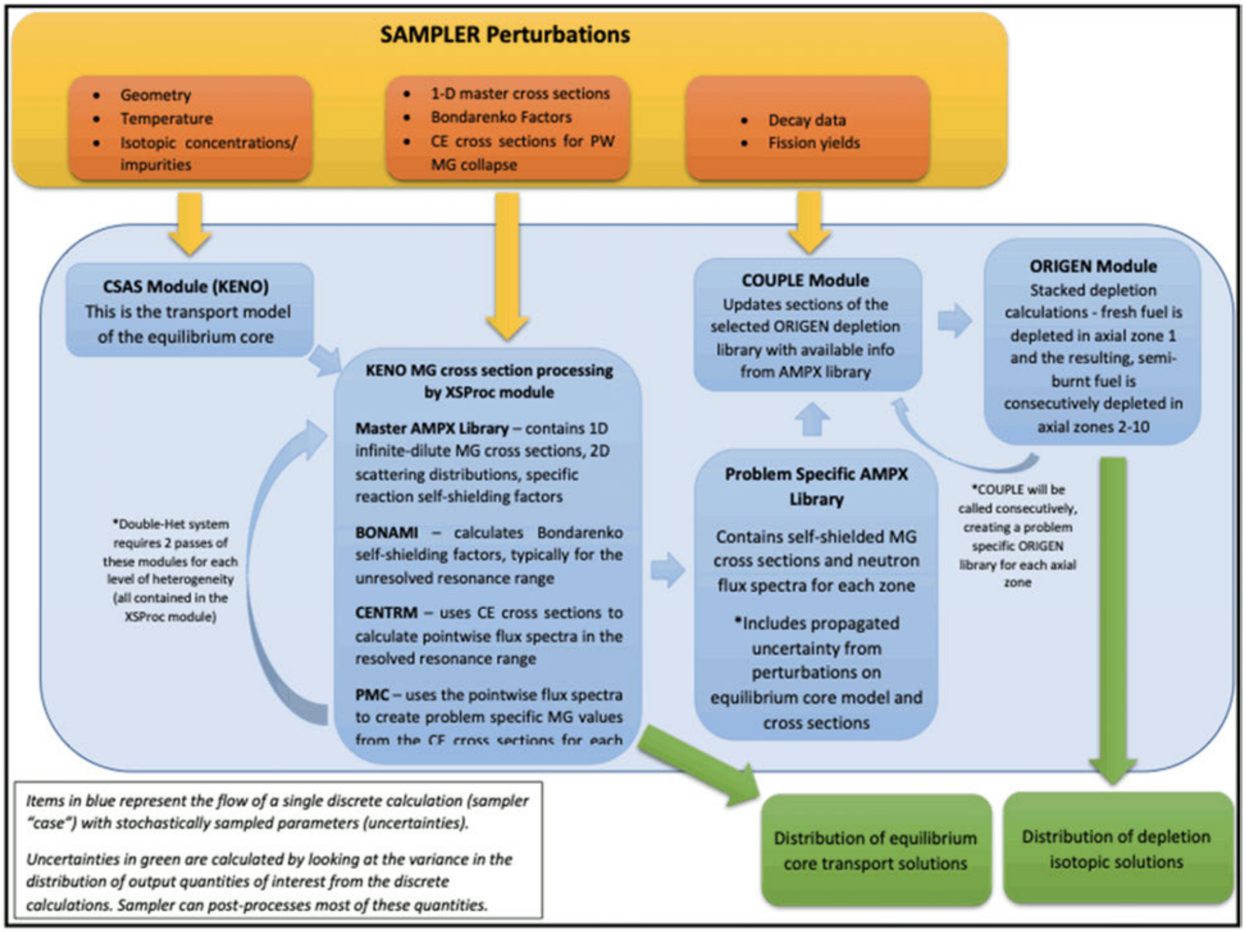

Figure 6-2. Depletion Methodology Flow Diagram for Burnup Calculations of Fuel Pebbles

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APPENDIX A EXAMPLE CORE DESIGN MODEL

This section presents an example of application of the core design modeling methodology for the evaluation of a KP-FHR reactor with a 35MWth power level. Figure A-1 shows a typical sequence of calculations performed by using the core design methodology described in this document.

Table A-1 summarizes the main Hermes core design input parameters considered in this example of application of core design methodology.

Figure A-2 shows the core geometry and nomenclature associated with core main regions.

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Table A-1. Core Design Input Parameters

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Table A-2. Zone-based Power Density Distribution

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Table A-3. Reactivity Coefficients at Startup and Equilibrium Core

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Table A-4. Reactivity Control System Requirements for Short-term Hot Shutdown

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Table A-5. Reactivity Shutdown System Requirements for Safe Shutdown

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Table A-6. Kinetic Parameters at Equilibrium Conditions

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Table A-7. Group-wise Effective Delayed Neutron Fraction and Corresponding Decay Constant at Equilibrium Core Conditions

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Table A-8. Kinetic Parameters at Startup Core Conditions

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Table A-9. Group-wise Effective Delayed Neutron Fraction and Corresponding Decay Constant at Startup Core Conditions

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Table A-10. Coolant Temperature Reactivity Coefficients for Flibe of Different Compositions

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Table A-11. k-eff with and without TH Feedback

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Figure A-1. Core Design Calculation Diagram

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Figure A-2. KP-FHR Core Geometry

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Figure A-3. Cross-sectional Views of Normalized Instantaneous Pebble Residence Time

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Figure A-4. Spectral Zones used for the Hermes Core

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Figure A-5. Fast (> 0.1 MeV) (left), Intermediate (middle), and Thermal (< 1.86 eV) (right) Neutron Flux in Hermes Equilibrium Core

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Figure A-6. Differential Worth of a Single Element Withdrawal, from All In (RCS only)

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Figure A-7. Reactivity Shutdown System Worth Curves, N-1

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Figure A-8. Power density (left), Flibe temperature (center), and Fuel Kernel Centerline Temperature (right)

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Figure A-9. Axial Binned Power Density Profile in the core, excluding Converging and Diverging Regions (left), and the Relative Difference of Axial Power Shape between Constant Temperature and KPATH Results (right)

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Figure A-10. Radial Binned Power Density Profile in the Core (left), and Relative Difference of Radial Power Shape between Constant Temperature and KPATH Results (right)

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APPENDIX B NEUTRONICS PIRT FOR THE KP-FHR

A Phenomena Identification Ranking (PIRT) evaluation was conducted for the KP-FHR. A summary of the results of the PIRT are included for information in this Appendix. The description of Figures of Merit (FOMs) and knowledge level numbering are as follows:

- FOM 1: Multiplication factor (1: **L**ow impact, 2: **M**edium impact, and 3: **H**igh impact)
- FOM 2: Power distribution (1: **L**ow impact, 2: **M**edium impact, and 3: **H**igh impact)
- Knowledge: Knowledge level (1: **L**ow, 2: **M**edium, and 3: **H**igh)

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