

**Enclosure 2**

**Fuel Qualification Methodology for the  
Kairos Power Fluoride Salt-Cooled High Temperature Reactor (KP-FHR)  
Revision 2**

**(Non-Proprietary)**



707 W. Tower Ave  
Alameda, CA 94501

# **Fuel Qualification Methodology for the Kairos Power Fluoride Salt-Cooled High Temperature Reactor (KP- FHR)**

## **Topical Report**

Revision No. 2  
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Non-Proprietary

Fuel Qualification Methodology for the Kairos Power Fluoride Salt-Cooled High Temperature Reactor (KP-FHR)			
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## Executive Summary

This document describes the fuel qualification methodology for the Kairos Power Fluoride Salt-Cooled High Temperature Reactor (KP-FHR). The fuel for the KP-FHR is comprised of Tristructural Isotropic (TRISO) fuel particles embedded in a carbon matrix spherical fuel pebble.

The foundation for the fuel qualification methodology of the KP-FHR fuel is international and U.S. operating experience with TRISO fuel particles and the extensive post-irradiation examination (PIE) and safety testing of TRISO fuel particles by the Department of Energy (DOE) as part of the Advanced Gas Reactor (AGR) fuel development experiments AGR-1 and AGR-2 (Reference 12). Although TRISO particle design and behavior is well understood based on U.S. and international experience, the specific fuel pebble design for the KP-FHR does not have an existing experience base. The fuel qualification methodology for the KP-FHR fuel pebble design is informed by the development of a Phenomena Identification and Ranking Table (PIRT) evaluation which identifies the important phenomena to be considered in establishing the fuel qualification envelope. The methodology includes development of a KP-FHR fuel specification that ensures the fuel manufactured for use in the KP-FHR remains within the fuel qualification envelope. The fuel qualification envelope defines the bounds of acceptable KP-FHR operation to ensure the qualification adequately covers the expected normal operation and licensing basis events (LBE).

The KP-FHR fuel qualification methodology includes consideration of mechanical/structural performance, chemical, thermal, and irradiation effects. Mechanical performance effects are confirmed by Laboratory Testing of the KP-FHR fuel pebble design and include mechanical, tribology, buoyancy, and material compatibility testing to ensure that the fuel pebble will retain its form and function during normal operation and LBEs. The methodology for confirmation of thermal and irradiation effects is based on testing performed in the AGR program, which defines an operational envelope for which the fuel particle is qualified. If the fuel is to be operated outside of this envelope, additional irradiation testing is performed to confirm that the fuel designed and manufactured for the KP-FHR will perform as intended under the conditions of this extended envelope during normal operation and postulated events. The KP-FHR uses a continuous pebble circulation and refueling concept and individual fuel pebbles pass through the core multiple times. Circulated pebbles are inspected in a pebble handling and storage system (PHSS) after each pass through the core which permits confirmatory monitoring of fuel pebble performance during the initial KP-FHR reactor startup and during equilibrium core operations. The confirmatory monitoring can detect defective fuel and also includes the measurement of radioactive noble-gas fission products in the reactor cover gas system and soluble fission products in the primary coolant chemistry control system.

The KP-FHR fuel qualification methodology provides reasonable assurance that the KP-FHR fuel pebble design can operate with the continued low TRISO particle failure fractions observed in the AGR testing program. The results of the implementation of the fuel qualification methodology will be submitted with applicable licensing applications submitted under 10 CFR 50 or 10 CFR 52. This report requests that the NRC approve the Sections 3.6 and 3.7 and the acceptance criteria in Table 4-1 of this report for performing qualification of KP-FHR fuel.

This methodology applies to a non-power test KP-FHR or a commercial electric power KP-FHR.

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## ABBREVIATIONS

Abbreviation or Acronym	Definition
AGR	Advanced gas reactor
ASTM	American Society for Testing and Materials
AOO	Anticipated operating occurrence
AVR	Arbeitsgemeinschaft Versuchsreaktor
BDBE	Beyond design basis event
CFR	Code of Federal Regulations
CO	Carbon monoxide
DBE	Design Basis Event
DLBL	Deconsolidation leach burn leach
DOE	Department of Energy
DUF	Dispersed Uranium Fraction
EKF	Exposed kernel fraction
FHR	Fluoride Salt-Cooled High Temperature Reactor
FIMA	Fissions per initial heavy metal atom
HALEU	High assay low enriched uranium
HFIR	High Flux Isotope Reactor
HFR	High flux reactor
HTGR	High temperature gas-cooled reactor
HTR-PM	High-Temperature Gas-Cooled Reactor Pebble-Bed Module
HTR-10	10MW High Temperature Gas-cooled reactor-test Module
HTTR	High-Temperature Engineering Test Reactor
IAEA	International Atomic Energy Agency
INL	Idaho National Laboratory
IPyC	Inner pyrolytic carbon
IRP	Integrated research project
KP-FHR	Kairos Power Fluoride Salt-Cooled High Temperature Reactor
LEU	Low enriched uranium
LWR	Light water reactor

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MHTGR	Modular High Temperature Gas Reactor
MSRE	Molten Salt Reactor Experiment
NDE	Non-destructive examination
NO	Normal operation
NRC	Nuclear Regulatory Commission
OPyC	Outer pyrolytic carbon
ORNL	Oak Ridge National Laboratory
PC&P	Process control and parameters
PDC	Principal design criteria
PHSS	Pebble handling and storage system
PIE	Post-irradiation examination
PIRT	Phenomena identification and ranking table
PRA	Probabilistic risk assessment
PyC	Pyrolytic carbon (Pyrocarbon)
QA	Quality assurance
RG	Regulatory guide
SARRDL	Specified acceptable system radionuclide release design limit
SiC	Silicon carbide
THTR	Thorium high-temperature reactor
TRISO	Tristructural isotropic
UCB	Upper confidence bound
UCO	Uranium oxycarbide (a mixture of UC, UC <sub>2</sub> , and UO <sub>2</sub> )

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## 1 INTRODUCTION

Kairos Power LLC (Kairos Power) is pursuing the design, licensing, and deployment of the Kairos Power Fluoride Salt-Cooled High Temperature Reactor (KP-FHR). One of the key features of this design is the use of TRISO fuel particles, each of which contains four barriers to radionuclide transport; these four barriers are a key component of the KP-FHR functional containment.

This report provides the Kairos Power methodology for the qualification of the KP-FHR fuel. This qualification methodology is based on existing qualification testing experience of TRISO fuel particles, performed by the DOE, and consideration of mechanical, thermal, chemical, and irradiation effects specific to the KP-FHR fuel pebble design. This methodology is described in detail in Section 3.

### 1.1 DESIGN FEATURES

#### 1.1.1 Design Background

To facilitate Nuclear Regulatory Commission (NRC) review and approval of this report for use by future applicants, key design features are provided in Section 1.1.2 which are inherent to the KP-FHR technology. These features are not expected to change during the design development by Kairos Power and provide the basis to support the safety review of this report. Should fundamental changes occur to these key design features, or new or revised regulations be promulgated that affect the KP-FHR fuel, such changes would be reconciled and addressed in license application submittals.

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

Although not intended to support the findings necessary to approve this report, additional design description information is provided in the technical report “Design Overview of the Kairos Power Fluoride Salt-Cooled, High Temperature Reactor” (Reference 2).

#### 1.1.2 Key Design Features of the KP-FHR

The KP-FHR is a high temperature reactor with molten fluoride salt coolant operating at near-atmospheric pressure. The fuel in the KP-FHR is based on the TRISO high-temperature coated particle fuel (originally developed for High-Temperature Gas-Cooled Reactors) in a spherical carbon matrix pebble. Coatings applied to the fuel kernel provide retention of fission products. The reactor coolant is a chemically stable molten fluoride salt mixture,  $2\text{LiF}:\text{BeF}_2$  (Flibe) which also provides retention of fission products that escape from any fuel defects. The KP-FHR includes a Pebble Handling and Storage System (PHSS) which extracts fuel from the core and based on inspection, either inserts it back into the active core or directs the fuel to spent fuel storage. While outside the active core, the fuel is not critical and produces only decay heat. Above the active core the fuel passes through a defueling chute, where short-lived fission products decay. The fuel in the PHSS is not covered by salt and the heat generation is low because short-lived fission

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products have decayed while traveling through the defueling chute. Fuel pebbles pass through the core multiple times before reaching full burnup. Fuel pebbles are inspected after each pass including a measurement of burnup. This measurement is used to determine if a pebble is to be moved to spent fuel storage and replaced with a fresh fuel pebble. The pebbles are also inspected for physical damage and replaced if necessary. Argon cover gas in the reactor vessel and in the PHSS is monitored for noble-gas fission products to assist in detection of defective fuel particles. In the PHSS, the functional containment is met by the layers of the TRISO fuel particle. The PHSS is described further in Section 2.2.5

A primary coolant loop circulates the reactor coolant using pumps and transfers the heat to a heat exchanger. The KP-FHR design includes two decay heat removal systems. A normal decay heat removal system is used following normal shutdowns and anticipated operational occurrences (AOOs). A separate passive decay heat removal system, along with natural circulation in the reactor vessel, removes decay heat in response to a design basis accident (DBA) and does not rely on electrical power. A schematic of the KP-FHR reactor is shown in Figure 1-1.

The KP-FHR design uses a functional containment approach similar to the Modular High Temperature Gas-cooled Reactor (MHTGR) instead of the typical light water reactor (LWR) low-leakage, pressure retaining containment structure. The KP-FHR functional containment design objective is to meet 10 CFR 50.34 (10 CFR 52.79) offsite dose requirements at the plant's exclusion area boundary with margin. A functional containment is defined in Regulatory Guide (RG) 1.232 as a "barrier, or set of barriers taken together, that effectively limit the physical transport and release of radionuclides to the environment across a full range of normal operating conditions, AOOs, and accident conditions." RG 1.232 includes an example design criterion for the functional containment (MHTGR Criterion 16). As also stated in RG 1.232, the NRC has reviewed the functional containment concept and found it "generally acceptable," provided that "appropriate performance requirements and criteria" are developed. The NRC staff has developed a proposed methodology for establishing functional containment performance criteria for non-LWRs, which is presented in SECY-18-0096. This SECY document has been approved by the Commission.

The functional containment approach for the KP-FHR is to control radionuclides primarily at their source within the coated fuel particle under normal operations and LBEs without requiring active design features or operator actions. The functional containment approach has also been proposed for high-pressure MHTGRs. Like the MHTGR, the KP-FHR design relies primarily on the multiple barriers within the TRISO fuel particles to ensure that the dose at the site boundary as a consequence of postulated accidents meets regulatory limits. However, in contrast to the MHTGR, the low-volatility KP-FHR molten salt coolant also serves as a distinct additional barrier providing retention of fission products that escape the fuel pebble barriers. This is due to the capability of the low-pressure molten salt coolant to absorb and immobilize solid fission products. Monitoring of soluble solid fission products in the primary coolant provides additional capability to detect defective fuel. This differs from HTGRs where fission products released from fuel absorb onto graphite dust and other surfaces that are challenging to monitor. The additional retention provided by the molten salt coolant is a key feature of the enhanced safety and reduced source term in the KP-FHR.

The KP-FHR fuel design consists of TRISO-coated particles embedded in a spherical pebble to form the fuel (Figure 2-1). The pebbles contain a central sub-dense core (inner core) surrounded by an annular layer of TRISO particles packed into partially graphitized matrix material (fuel annulus) and covered by an outer shell of fuel-free matrix material. Each TRISO particle is comprised of a fissile uranium oxycarbide (UCO) kernel surrounded by a porous carbon buffer layer and coated by a silicon carbide (SiC) layer sandwiched

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between two dense pyrolytic carbon (PyC) layers. The TRISO particles are overcoated with resinated graphite powder before being pressed into the fuel annulus. The resinated graphite powder transforms into carbon matrix material upon pressing and subsequent heat treatments. Table 1-1 provides a high-level summary of the KP-FHR design.

## 1.2 REGULATORY REVIEW

The TRISO fuel design provides four of the five credited fission product barriers to the release of radioactivity in the KP-FHR, which comprise the functional containment. These barriers are the fuel kernel itself, the inner pyrolytic carbon (IPyC) layer, the silicon carbide (SiC) layer, and the outer pyrolytic carbon (OPyC) layer. The final barrier credited in the KP-FHR is the molten salt coolant.

The Kairos Power licensing framework is contained in two topical reports, “Principal Design Criteria for the Kairos Power Fluoride Salt-Cooled High Temperature Reactor” (Reference 3) and “Regulatory Analysis for the Kairos Power Fluoride Salt-Cooled, High Temperature Reactor” (Reference 4). The NRC regulations and Principal Design Criteria (PDC) that are relevant to the fuel design are described below.

### 1.2.1 Regulations Relevant to the KP-FHR Fuel Qualification

The regulations that are relevant to the fuel are contained in 10 CFR 50.34 “Contents of applications; technical information”, 10 CFR 50.43(e) “Additional standards and provisions affecting class 103 licenses and certifications for commercial power,” and 10 CFR 100.10, “Factors to be considered when evaluating sites.”

#### 10 CFR 50.34

10 CFR 50.34(a) is relevant to the requirement to describe design characteristics of the KP-FHR fuel, Flibe and all supporting structures, systems, and components in order to include a preliminary safety analysis report with the application for a construction permit.

*50.34(a)(1)(ii)(C) “The extent to which the reactor incorporates unique, unusual or enhanced safety features having a significant bearing on the probability or consequences of accidental release of radioactive materials.”*

*50.34(a)(1)(ii)(D) “The safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur. Special attention must be directed to plant design features intended to mitigate the radiological consequences of accidents.”*

*50.34(a)(2) “A summary description and discussion of the facility, with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations.”*

#### 10 CFR 50.43(e)

*This regulation requires that designs which differ significantly from light-water designs licensed before 1997 will be approved only if:*

*(1)(i) The performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience, or a combination thereof;*

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*(ii) Interdependent effects among the safety features of the design are acceptable, as demonstrated by analysis, appropriate test programs, experience, or a combination thereof; and*

*(iii) Sufficient data exist on the safety features of the design to assess the analytical tools used for safety analyses over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions; or*

*(2) There has been acceptable testing of a prototype plant over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions. If a prototype plant is used to comply with the testing requirements, then the NRC may impose additional requirements on siting, safety features, or operational conditions for the prototype plant to protect the public and the plant staff from the possible consequences of accidents during the testing period.*

#### 10 CFR 100.10

Subsections within 10CFR100.100(a) are relevant to the requirement to describe how the KP-FHR fuel and FLiBe provide protection against the hazardous consequences of an accident and thus ensure a low risk of public exposure.

#### 10CFR100.10(a)(3)

*“Characteristics of reactor design and proposed operation including the extent to which the reactor incorporates unique or unusual features having a significant bearing on the probability or consequences of accidental release of radioactive materials.”*

The KP-FHR fuel contains fission product barriers credited for mitigating the release of radioactivity during AOs and design basis events (DBEs). The KP-FHR TRISO based fuel is considered to represent a new or unique feature which has a significant bearing on the consequences of the release of radioactive materials. Kairos Power is requesting NRC review and approval that the fuel qualification methodology described in this report is sufficient to demonstrate the fission product retention capabilities of the KP-FHR fuel.

### **1.2.2 Principal Design Criteria that are Relevant to KP-FHR Fuel Qualification**

The following PDC are relevant to the qualification of the KP-FHR fuel:

PDC 10:

*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 16:

*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.*

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The design and qualification of the KP-FHR fuel is such that the Specified Acceptable System Radionuclide Release Design Limit (SARRDL) is met during any conditions of normal operation and AOOs.

Coatings on the TRISO fuel particle provide retention of fission products well in excess of those expected in a KP-FHR during normal operation, AOOs, and DBEs; and therefore, act as a functional containment, as defined RG 1.232. The reactor coolant also provides retention of fission products that escape from any fuel defects. The performance of fission product retention in the reactor coolant is outside the scope of this report. Qualification of the KP-FHR fuel design will confirm the pebble fuel element meets this fission product containment function.

### 1.2.3 Regulatory Request

Kairos is requesting NRC approval of the fuel qualification methodology described in Section 3, specifically Sections 3.6 and 3.7 and Table 4-1, “Acceptance Criteria for KP-FHR Initial Fuel Qualification.” This qualification methodology is based on a combination of analysis, testing, and operating experience to meet the requirements of 10 CFR 50.34(a)(1)(ii)(C), 10 CFR 50.43 (e)(1), 10CFR100.10(a)(3), PDC 10, and PDC 16. Kairos Power requests that the NRC review and approve this methodology and expects that, when the acceptance criteria for the approved methodology in Table 4-1 are met, the fuel described in this report is qualified for use by applicants for licenses of a KP-FHR under 10 CFR 50 and 10 CFR 52. This methodology applies to fuel qualification for a non-power test or commercial electric power KP-FHR.

### 1.2.4 Other Fuel-Related Topical and Technical Reports

This topical report provides the methodology to qualify the fuel for use in a KP-FHR. The fuel design details and the results of the fuel qualification will be addressed in safety analysis report documents provided as part of licensing applications submitted under 10 CFR 50 or 10 CFR 52.

Other fuel related topical reports include the “KP-FHR Fuel Performance Methodology” (Reference 19), “Source Term Methodology” (Reference 41), and “KP-FHR Risk-Informed Performance-Based Licensing Basis Development Methodology,” (Reference 23). Related technical reports include the “KP-FHR Core Design and Analysis Methods” (Reference 42) and the “Postulated Event Methodology” (Reference 43).

The Fuel Performance topical report provides the methodology which is used to perform fuel performance design studies and also provides fuel failure fraction and fission product release fractions during normal operation and licensing basis events to be used in the source term analysis. The Source Term topical report provides the methodology for determining the radiological source term. The Core Design technical report provides the methodology for performing neutronic and thermal hydraulic analysis of the core and provides power, coolant temperature, burnup, and fluence and their associated uncertainties for use in fuel performance analysis. The risk-informed performance-based licensing basis development methodology identifies the licensing basis events that are analyzed as part of the safety analysis report. The postulated event technical report describes the methodology for analyzing licensing basis events and provides transient information to the fuel performance code for analysis of failure fractions and fission product release.

The interrelationships between the methodologies are shown in Figure 4-2.

## 2 FUEL DESIGN



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Fuel qualification provides high confidence that the physical, chemical, and irradiation behavior of fuel is sufficiently understood so that the fuel can be safely operated and adequately modeled for both normal and LBE conditions, reflecting the role of the fuel design in the overall safety of the facility. It also provides high confidence that residual uncertainties are understood such that any calculated fission product releases from the fuel include appropriate margin to ensure conservative calculation of radiological dose consequences.

This section provides a description of the fuel pebble design and background for its development. In addition, details on the particle and pebble design, background information for the core design, and the pebble handling system are provided to aid in the understanding of the fuel qualification methodology described in Section 3. Any departures from the fuel pebble design described in this section that impact the fuel qualification methodology will be addressed and justified in safety analysis report documents provided as part of licensing applications submitted under 10 CFR 50 or 10 CFR 52.

## 2.1 BACKGROUND OF THE KP-FHR FUEL DESIGN

The KP-FHR utilizes an annular fuel pebble design with characteristics specific to application in FHRs. The design of the annular pebble is similar to the traditional German pebble design used in pebble bed gas-cooled reactors that was developed in the 1960s and improved in the 1970s and 1980s. The purpose of this background discussion is to provide a basic understanding of the similarities and differences in the designs, and to provide the reasoning behind the design of the KP-FHR annular pebble. A comparison of the KP-FHR fuel pebble and the traditional fuel pebble design is provided in Table 2-1.

The traditional fuel pebble design was effectively implemented in the German pebble-bed HTGR program. This pebble design is 60mm in diameter and has two regions; a 50mm diameter spherical carbon matrix fuel region containing a 7-12% packing fraction of BISO<sup>1</sup> or TRISO particles and a 5mm thick fuel-free outer shell to protect the fuel region. The fissile TRISO fuel particles in this pebble design contained a range of fuel kernel types<sup>2</sup>. This TRISO fuel particle design was extensively tested in test reactors and effectively used in the AVR German reactor (Arbeitsgemeinschaft Versuchsreaktor), and BISO fuel particles were used in the THTR (Thorium High-Temperature Reactor) (References 5 and 6). The Chinese pebble-bed reactor program used this same pebble design with low enriched UO<sub>2</sub> TRISO particle in a prototype reactor, the 10 MW High Temperature Gas-Cooled Reactor-Test Module (HTR-10), and is used in the High-Temperature Gas-Cooled Reactor Pebble-Ped Module (HTR-PM) demonstration plant that may achieve commercial operations this year. These programs are discussed in References 7, 8, 9, 10.

The KP-FHR fuel pebble design is considered to be an improved version of the gas reactor fuel design that is unique in its own right, but also has a geometry, irradiation environment, and uses materials that are similar to traditional HTGR fuel pebbles. This similarity allows for consideration of the existing experience base of TRISO fuel particles and pebble fuel elements from U.S. domestic and international programs in Germany, UK, Japan, France, Russia, South Africa, South Korea, and China in the qualification methodology for the KP-FHR pebble design. All of these programs developed a manufacturing process for TRISO fuel particles and carbon-based fuel elements in either spherical pebble or cylindrical compact forms. The

<sup>1</sup> BISO particles consisted of a high density kernel coated with a low density pyrocarbon buffer layer and a high density pyrocarbon outer layer

<sup>2</sup> (HEU, Th)C<sub>2</sub>, (HEU, Th)O<sub>2</sub>, LEUO<sub>2</sub>, HEUO<sub>2</sub>, ThO<sub>2</sub>, HEUC<sub>2</sub>, HEUCO, LEUO<sub>2</sub>

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performance of fuel manufactured in these programs was characterized through irradiation and safety testing and with subsequent use in prototype HTGRs in the US, Germany, UK, Japan, and China.

## 2.2 KP-FHR FUEL PEBBLE DESIGN DESCRIPTION

### 2.2.1 Particle Design

The KP-FHR TRISO particles include a UCO kernel, a porous carbon buffer layer, and IPyC, SiC, and OPyC layers. TRISO particles are overcoated with a mixture of natural and synthetic graphite, and resin binder material. 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. The function and dimension of each component of the TRISO fuel particle are provided in Table 2-2. The particle design is similar to the particles tested in the second campaign of the AGR program, AGR-2. Table 2-2 summarizes the particle components, their dimensions, and the purpose of each component.

### 2.2.2 Pebble Design

The KP-FHR fuel pebble design is 40mm in diameter and has three regions with specific functions that complement the pebble-bed FHR design (Figure 2-1). The smaller diameter pebble (as compared with the traditional HTGR pebble design) results in a greater surface area for heat removal from the fuel pebble thereby reducing fuel temperature. The use of smaller pebbles is feasible because molten salt is a more effective thermal fluid in comparison to helium gas in an HTGR. The volumetric heat capacity of Flibe is more than two orders of magnitude greater than helium and the thermal conductivity is several times greater. This enables the core average power density of an FHR to be several times greater than an HTGR.

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. A fuel region shell is located on the surface of the inner carbon matrix core. The fuel annular 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 reduces the fuel temperatures. The TRISO particles used for the KP-FHR fuel design have a fuel specification similar to the DOE AGR Fuel Development and Qualification Program fuel particles (References 11 and 12). Critical parameters related to fuel performance are derived from the AGR specification. The KP-FHR fuel particle kernels are composed of UCO, a mixture of  $\text{UO}_2$ , UC, and  $\text{UC}_2$  which has been produced, irradiated and safety tested in the U.S. in multiple experiments since the late 1970s. The addition of carbon to the kernel mitigates the generation of carbon monoxide (CO) gas. This reduces the risk of kernel migration, over-pressurization of the particle with CO gas in addition to fission product gases, and CO gas reactions with the SiC layer. 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.

### 2.2.3 KP-FHR Core Design Strategy

The KP-FHR core contains thousands of pebbles that circulate through the core. The reactor core will transition from a startup core and evolve to a higher performance equilibrium core over time. The ability to circulate different pebble enrichments through the core allows the composition of the individual pebbles in the pebble bed to be altered slowly enabling these changes in core performance. The KP-FHR fuel pebbles may contain enriched uranium-235 up to 19.55 wt% and natural uranium to reduce effective enrichment and core reactivity in early startup core operations. The core also contains graphite pebbles in a ratio to fuel pebbles depending on the chosen startup and operation option. The ability to control the

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variation in pebble type in the core allows excess reactivity to be minimized during startup and operations. During the initial startup, fission products will accumulate, requiring an increase in the number of enriched fuel pebbles. Higher enrichment pebbles may also be added to the core to maintain a minimum level of reactivity. The design and analysis of the core is outside the scope of this report and will be addressed in a separate topical report for core design and analysis.

Another core design factor is the core carbon to heavy metal (C/HM) atom ratio that depends on the fuel pebble design and types of pebbles in the core. This ratio describes the level of moderator present and is a key parameter that influences core behavior. The core can safely operate within a specified C/HM range before reaching an undesirable level of under-moderation or over-moderation. A core design with a high C/HM ratio is considered optimal for maximizing fuel utilization.

The core is continuously fueled by the PHSS adding pebbles at the bottom of the core and removing them at the top of the core. The pebbles are designed to be buoyant and move slowly up through the core. Pebble transit time varies depending on recirculation rates and pebble location in the core, and the coolant transit time is significantly faster. The coolant flows upward in the same direction as the fuel pebbles move through the core. When pebbles are removed, their physical condition and burnup are assessed by monitoring and inspection equipment in the PHSS. After passing inspection they are re-inserted into the core by the PHSS, which is described in Section 2.2.5.

A cover gas system is located at the top of the PHSS defueling chute, above the molten salt coolant. The cover gas system is designed to maintain an inert atmosphere in the reactor core by use of continuous circulation of argon gas over the free surface of the FLiBe. This cover gas removes tritium and other gases or particulates for further treatment. The cover gas system is monitored for the presence of radioactivity from fission products. In addition, the FLiBe in the primary system is monitored for activity. Both of these systems are early indicators of fuel failure and migration of fission products out of the fuel particles. The FLiBe is the final barrier credited to mitigate the release of fission products for design basis accident analyses.

#### 2.2.4 Fuel Definitions

To aid in the subsequent fuel qualification discussion, the definitions for some key terms are provided below:

TRISO Particle – contains the UCO kernel, buffer, IPyC, SiC, and OPyC coating layers.

Fuel Pebble – contains the TRISO particles in a fuel annulus, a lower density inner core, and a fuel free outer matrix shell.

Fuel Element – is the generic term for fuel, which includes compacts, pebbles, fuel rods. For the sections of this document discussing testing and other industry experience, the term fuel element is used. The term fuel pebble is used in the remainder of the document.

TRISO Failure – corresponds to the loss of integrity of all three outer coating layers and is detected by the release of fission gases.

SiC Failure – corresponds to the loss of integrity of the SiC layer, the primary barrier to the release of fission products, with at least one remaining intact PyC layer such that fission gases are retained in the TRISO particle. SiC failure is usually detected by the release of cesium.

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The fuel particle and pebble are shown in Figure 2-1.

### 2.2.5 Pebble Handling and Storage System

The PHSS removes pebbles after they have transited the core, inspects them, and then reinserts them at the bottom of the core. While they are removed from the core, the pebbles are inspected for burnup and for the presence of physical degradation. The burnup is determined by a gamma ray spectrometer and physical degradation is assessed by means of inspection. If the pebble has achieved or is nearing its design burnup or shows evidence of physical degradation, it is removed from service. Pebbles are available for inspection multiple times in a pebble lifetime. As a result, there are numerous opportunities to identify issues with the fuel well before reaching the burnups that will be achieved in the high burnup equilibrium core.

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### 3 FUEL QUALIFICATION METHODOLOGY

#### 3.1 OVERVIEW

Fuel qualification includes consideration of mechanical, chemical, thermal, and irradiation effects on the fuel design consistent with operating conditions. The qualification for the KP-FHR fuel pebble relies significantly upon the existing U.S. and international historical experience with the TRISO particle and pebble fuel elements and the advancement in fuel technology developed as part of the U.S. DOE AGR Fuel Development and Qualification Program (AGR fuel program). The U.S. DOE initiated the AGR fuel program in the early 2000s to develop a UCO TRISO fuel manufacturing process, demonstrate TRISO fuel performance, and provide data for fuel qualification in support of reactor licensing. A critical part of this effort was evaluating past issues with U.S.-manufactured particle fuel in comparison to the successful German experience. This effort resulted in a TRISO fuel particle design that was first fabricated at a laboratory scale and, subsequently, at an engineering scale and then irradiated in a series of tests in the Advanced Test Reactor (ATR) at Idaho National Laboratory (INL). The first two irradiation test campaigns, AGR-1 and AGR-2, serve as the foundation for the qualification of a TRISO fuel particle design for application in the KP-FHR.

Although TRISO particle design and behavior is well understood based on U.S. and international experience, the specific fuel pebble design for the KP-FHR does not have an existing experience base. A fuel pebble phenomena identification and ranking table (PIRT) exercise was performed on the KP-FHR fuel pebble design by Kairos Power along with industry subject matter experts. The purpose of the PIRT was to evaluate fuel pebble phenomena against the fission product transport and release from the fuel to the coolant for given scenarios of manufacturing, normal operations, AOOs, and LBEs. For the KP-FHR, the figure of merit has been determined to be the SARRDL, as discussed in the KP-FHR PDC topical report (Reference 3). The KP-FHR fuel pebble PIRT builds upon the Next Generation Nuclear Plant fuel PIRT conducted between 2002 and 2004 for the NRC (References 20, 21, 22). This existing PIRT was updated for the KP-FHR based on the application of TRISO fuel particles to the fuel pebble in the KP-FHR environment and by incorporating knowledge gained on fuel performance from the AGR program since 2004. The KP-FHR PIRT identified areas where additional analysis or testing of the KP-FHR fuel pebble design was warranted for certain phenomena with medium and high importance ratings when evaluated relative to the figure of merit. The KP-FHR PIRT conclusions inform the fuel qualification methodology, including in particular the laboratory and irradiation test programs on the KP-FHR fuel pebble design that are discussed in this section.

The AGR test experience provides confidence that TRISO particle failure fractions in particles manufactured to the same specification in a quality-controlled program will result in similar very low failure fractions under similar irradiation conditions. The KP-FHR fuel particle design and specification is derived considering the specifications for AGR fuel particles for all critical parameters with the expectation that a consistent fuel particle design will behave as observed in the AGR irradiation test program. The Kairos Power fuel pebble manufacturing process will be developed with input from industry experts.

A fuel operating envelope defining the range of conditions for which the fuel is to be qualified is developed based on analyses of normal operating and LBE conditions. These analyses will identify the most limiting conditions the fuel will be expected to experience. This envelope is compared to the fuel qualification envelope to ensure that the qualification covers the full range of operating and LBE conditions.

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As noted above, the qualification for the KP-FHR fuel particle relies principally on the existing U.S. and international historical experience. Qualification of the KP-FHR fuel pebble will include laboratory testing and irradiation testing programs summarized below.

A laboratory test program will be conducted on the KP-FHR fuel pebble to demonstrate acceptability of the annular fuel pebble form with respect to pebble mechanical integrity, buoyancy, and material compatibility with Flibe coolant. These tests will be performed with surrogate fuel kernels and include mechanical testing, molten-salt and inert gas tribology, buoyancy tests, and material compatibility testing of the pebble in Flibe and gas environments. These laboratory tests will also include compression, impact, and tribology tests. Buoyancy testing will examine the propensity for salt to infiltrate into pebbles and confirm the ability of the pebble to maintain net positive buoyancy under normal and off normal conditions. Flibe compatibility testing examines the interaction of molten salt with the pebble carbon matrix and TRISO fuel particles.

If the fuel operating envelope extends beyond the fuel qualification bounds of AGR irradiations, additional irradiation testing of KP-FHR fuel pebbles will be conducted in a non-KP-FHR test facility to ensure the fuel is qualified for these operational limits. This irradiation testing of KP-FHR fuel will collect fuel performance data at burnup, temperature, fast fluence and power conditions expected to be experienced in the core. This data will determine the fuel particle failure fraction for irradiation test specimens and will be used to confirm the pebble form does not alter the performance of TRISO particle from the behavior observed in the AGR program.

In addition to pre-operational qualification activities, a fuel surveillance program will be implemented in the KP-FHR to confirm that fuel performance behavior remains within design requirements during its service life in the reactor. This is an ongoing program that will monitor the fuel during startup, initial operations, and full power equilibrium operation. This program will monitor the cover gas and Flibe coolant to detect abnormal fission product releases from the fuel. Each fuel pebble will undergo a non-destructive examination (NDE) in the pebble handling system after each pass through the core. These NDE inspections include measurements of burnup with a gamma ray spectrometer and detection of damage by inspection. Destructive post irradiation examinations will also be performed on a set of pebbles at peak burnup to confirm fuel performance for the initial KP-FHR as discussed in Section 3.9.3.

The fuel qualification methodology demonstrates that there is reasonable assurance of safe operation of the KP-FHR fuel.

The remainder of this section discusses each of the elements in the fuel qualification methodology in more detail:

- U.S. and International Experience
- Fuel Pebble PIRT
- Fuel Specification, Manufacturing, and Quality Control Through Inspection
- Fuel Operating Envelope
- Fuel Pebble Laboratory Testing
- Fuel Irradiation Testing
- Fuel Performance Modeling

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- Fuel Surveillance

In summary, the U.S. and International Experience provides the primary qualification of the TRISO particle; the PIRT identifies areas where additional testing, analysis, or modeling is needed to further the understanding of fuel behavior; the Fuel Specification, Manufacturing, and Quality Control program ensures that the fuel particle manufactured for the KP-FHR is equivalent to that used in the AGR program; the Fuel Operating Envelope ensures that the fuel is operated within the qualification envelope; the Laboratory Testing Program addresses mechanical integrity of the pebble and items identified by the PIRT; the Fuel Irradiation Testing demonstrates that the particle is qualified for core operating conditions when operating conditions extend beyond the AGR irradiation test based qualification envelope; and finally the ongoing Fuel Surveillance Program confirms that the fuel is operating as designed. These actions collectively provide reasonable assurance that the fuel can be used safely in the KP-FHR.

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## 3.2 U.S. AND INTERNATIONAL EXPERIENCE

The foundation of the KP-FHR fuel qualification program is built on the U.S. and international experience with TRISO fuel particles over multiple decades and the U.S. DOE AGR program. The U.S. and international operating experience demonstrated the fundamental capability of the TRISO fuel particle. The AGR program manufactured TRISO fuel particles at an engineering scale and extensively tested the fuel at normal operating and accident conditions relevant to the KP-FHR. A summary of these programs is provided below and provides the basis for TRISO particle qualification for the KP-FHR.

### 3.2.1 TRISO Fuel Particle Manufacturing and Irradiation Testing

The first development and production of  $\text{UO}_2$  TRISO-coated particle fuel occurred in the UK in the early 1960s. The first prototype reactor incorporating  $\text{UO}_2$  TRISO fuel was the Dragon Reactor, which began operating in 1964 utilizing  $\text{UO}_2$  TRISO as the driver fuel with the capability of testing multiple fuel forms in test locations within the core. The reactor operated successfully as a fuel test facility until 1975, with high fission product retention within the driver fuel particles, as indicated by the following statement from a paper on the Dragon operating experience (Reference 27):

*“The low release of FP [fission products] to the primary circuit and the almost complete absence of corrosion not only improved the sensitivity of individual release measurements, but also allowed hands-on maintenance of non-activated components and personnel access to the containment building and during operation.”*

The German HTR development program demonstrated pebble-bed gas reactor technology with the small prototype AVR with a long operating history (1967-1988) (Reference 28). One large German pebble-bed commercial prototype, the THTR was developed in the 1970s, operated for a short period (1983 – 1986) with no follow-on development (Reference 5). BISO and TRISO fuel particles and fuel pebbles were developed for several advanced German pebble-bed reactor designs (mid-1970s thru 1988) and extensively tested in the “real-time” AVR and other European test reactors. Full commercial-scale production batches of high-quality fuel elements containing TRISO-coated fuel particles were irradiation-tested in the AVR (Reference 6). The performance of these pebbles was monitored and pebbles were sampled through an AVR surveillance program for post-irradiation examination (PIE) and accident testing. The fuel cycle concepts investigated included fissile and fertile fuel particles to promote the breeding and burning of uranium and thorium fuel kernels.

In the last fifteen years, TRISO testing in the High Flux Reactor (HFR) in Petten, The Netherlands, included archived German pebbles and pebbles fabricated in China (References 7 and 9). These tests demonstrated the performance of historical German-fabricated fuel compared with fuel fabricated by the Institute of Nuclear and New Energy Technology of Tsinghua University, China (References 14 and 15). These comparison tests were followed by a test with only fuel pebbles fabricated by the Chinese. These tests in the HFR were performed to support licensing and operation of the twin reactor 210 MWe HTR-PM in Shidao Bay, China (Reference 10). These reactors were built on the successful experience operating the smaller prototype HTR-10 at Tsinghua University (Reference 8) and testing of Chinese-fabricated pebbles in the Russian IVV-2M reactor (Reference 30). Additional TRISO fuel fabrication and testing was conducted in Russia and France.

Japan has conducted an extensive low-enriched  $\text{UO}_2$  TRISO development program leading to the construction and sustained operation of the prototype High-Temperature Engineering Test Reactor



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(HTTR) (Reference 31). The development and implementation of UO<sub>2</sub> TRISO fuel fabrication capability and production of fuel for the HTTR was conducted by the Japan Atomic Energy Research Institute (JAERI – now Japan Atomic Energy Agency, JAEA) in cooperation with Nuclear Fuel Industries, Ltd. (NFI) beginning in 1970 (Reference 32). The HTTR fuel was in the form of hexagonal graphite blocks with annular compacts in graphite sleeves placed in holes within the blocks. TRISO particles were produced with nine different enrichments ranging from 3.4 to 9.9% to control power distribution in the HTTR. The quality of the Japanese fuel was on a par with the German fuel and performance was demonstrated through multiple irradiations in the Japan Materials Testing Reactor, as well as one irradiation test in High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory (ORNL) to support the startup and operation of the HTTR (Reference 17).

The U.S. prototype and demonstration experience for gas reactors is from two General Atomics designed gas reactors, Peach Bottom Unit 1 in Pennsylvania and Fort St. Vrain in Colorado. These cores operated for several years and used prismatic blocks of graphite instead of pebble beds like the German and Chinese reactors. The first Peach Bottom core contained UC<sub>2</sub> kernels with a single dense pyrocarbon coating, the second core contained BISO coated UC<sub>2</sub> kernels. The Fort St. Vrain fuel included HEU/ThC<sub>2</sub> fissile particles and ThC<sub>2</sub> fertile particles. During this same time frame, U.S. DOE projects included the testing of TRISO fuel particles in compact form in HFIR at ORNL (Reference 16). These tests supported projects such as the New Production Reactor and efforts with General Atomics, Combustion Engineering, Bechtel and Stone & Webster to commercialize the MHTGR and the Gas Turbine Modular Helium Reactor concepts.

The last few TRISO irradiation tests in the 1990s took place in HFIR and included the TRISO-P fuel particle (which featured a thicker and denser IPyC layer and an added a porous “protective” PyC outer layer, named P-PyC) with a UCO fuel kernel. The irradiation performance in these tests resulted in higher particle failure fractions and fission product release than was observed in the German programs. It was determined that “as-fabricated fuel” parameters (IPyC anisotropy, protective PyC porosity and matrix intrusion) among other factors such as highly accelerated irradiation testing resulted in the observed performance difference between the U.S. and German programs (Reference 18). In the beginning of the AGR program, a review of the German and U.S. efforts led to improvements in the TRISO fuel particle design, manufacturing process, and the design of irradiation tests. These improvements resulted in the AGR program producing fuel with irradiation performance comparable to German fuel. The envelope of AGR testing (power, burnup, temperature, and fast fluence) exceeds that of the German fuel. The German, Chinese, Japanese, and U.S. legacy tests are summarized in Table 3-1 based on information from References 5, 6, 7, 9, and 16.

### 3.2.2 AGR-1 and AGR-2

The AGR Fuel Development and Qualification Program was initiated by the U.S. DOE in 2002 with the purpose of establishing the capability to fabricate high quality TRISO fuel (References 11 and 12). AGR-1 and AGR-2, the first two AGR fuel irradiation tests, included a total of 108 UCO fuel compacts that were irradiated at compact time-average maximum temperatures ranging from 1069 to 1360°C and burnups from 7.3 to 19.6% fissions per initial heavy metal atom (FIMA). The AGR-1 and AGR-2 manufacturing process also involved a level of intentional variation in fabrication equipment and processes. The kernels and coatings of the UCO particles tested in AGR-1 and AGR-2 also exhibited some degree of property

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variation and were fabricated under different conditions and at different scales (for example use of a 2-inch versus a 6-inch coater) with similar irradiation and accident safety performance.

The AGR-2 TRISO fuel particles have a kernel diameter of  $\sim 427\mu\text{m}$  meeting the KP-FHR fuel specification while the AGR-1 particles had a smaller kernel diameter of  $\sim 350\mu\text{m}$ . The small diameter AGR-1 TRISO particles achieved a peak burnup of 19.6%FIMA, which was higher than the peak burnup of 13.2%FIMA in AGR-2. A fuel performance metric was developed to assess different TRISO particle designs and to compare with the German TRISO fuel particle design as a reference point as described in the Electric Power Research Institute (EPRI) topical report on UCO TRISO Coated Particle Fuel Performance (Reference 13, Equation 4-1 and Table 4-2) The metric is based on the tangential tensile stress that develops in the SiC layer under internal gas pressure produced from gaseous fission products that accumulate with burnup.

The metric was normalized to the historic German value, recognizing the absence of pressure vessel failure in its  $\text{UO}_2$  fuel. The AGR-1 (350- $\mu\text{m}$ , 19.6% FIMA) and AGR-2 (427- $\mu\text{m}$ , 13.2% FIMA) have similar tensile SiC metric values than the German fuel ( $\sim 0.9$  vs 1). In AGR-1, the smaller kernel allows for a higher burnup. The KP-FHR fuel has a kernel diameter similar to AGR-2 but its projected maximum burnup is  $\sim 50\%$  higher. As a result, its tensile stress metric is about 30% higher than the German reference. The German reference was established without accounting for CO contribution to the internal pressure of its  $\text{UO}_2$  fuel and the influence of fuel temperature on gas pressure at the operating conditions of the reactor. Considering that CO is responsible for the majority of the internal pressure in  $\text{UO}_2$  fuel, this means that the well-performing German fuel has a true tensile stress metric value much higher than the reference value based on fission gas pressure only. Furthermore, fuel temperatures in an FHR are lower than typical high temperature gas reactors resulting in lower gas pressures in a TRISO fuel particle. Consequently, the KP-FHR fuel design margin to failure by overpressure is much larger than indicated by the metric. Despite the indications that AGR-2 type TRISO fuel particles can withstand burnup up to 19.6% FIMA, we are conservatively limiting the fuel qualification limit to 13.2% FIMA. This limit can be removed using the methodology in Section 3.7.

The AGR-1 and AGR-2 tests demonstrated that a high level of fuel performance and fission product retention was achieved within the bounds of the irradiation test. Testing results show that the radioactive source term emitted from the particle design is at least one order of magnitude lower than other reactor types. The irradiation test and PIE data from AGR-1 and AGR-2 will be used to address PIRT phenomenon (Tables 3-2 and 3-3) related to the irradiation performance of TRISO fuel particles in and in combination with the KP-BISON fuel performance model described in Section 3.8.

The results from the AGR-1 and AGR-2 tests are documented in the EPRI topical report (Reference 13), which makes the following conclusions in Section 8 of the topical report:

- “Testing of UCO TRISO-coated fuel particles in AGR-1 and AGR-2 constitutes a performance demonstration of these particle designs over a range of normal operating and off-normal accident conditions. Therefore, the testing provides a foundational basis for use of these particle designs in the fuel elements of TRISO-fueled HTR designs (i.e., designs with pebble or prismatic fuel and helium or salt coolant).”
- “The kernels and coatings of the UCO TRISO-coated fuel particles tested in AGR-1 and AGR-2 exhibited property variations and were fabricated under different conditions and at different scales, with remarkably similar excellent irradiation and accident safety performance results. The

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ranges of those variations in key characteristics of the kernels and coatings are reflected in measured particle layer properties provided in Table 5-5 (in the EPRI report). UCO TRISO-coated fuel particles that satisfy the parameter envelope defined by measured particle layer properties in Table 5-4 can be relied on to provide satisfactory performance.”

- “Aggregate AGR-1 and AGR-2 fission product release data and fuel failure fractions, as summarized in the EPRI report, can be used to support licensing of reactors employing UCO TRISO-coated fuel particles that satisfy the parameter envelope defined by measured particle layer properties in Table 5-5 from AGR-1 and AGR-2.”

The NRC safety evaluation report (SER) for the EPRI TRISO topical report is contained in Reference 36. This SER concludes there is reasonable assurance that TRISO particles produced to the specifications and limited to the performance parameters documented in the TRISO topical report will satisfy a portion of the requirements associated with PDC 10, subject to the Limitations and Conditions in Section 4.0 of the SER. The Kairos Power fuel qualification methodology will satisfy these limitations and conditions as described in Table 3-4.

The SER also discussed that the AGR fuel specification ranges were wider than the tested ranges. Kairos Power will use a fuel specification that is derived from and similar to the AGR-2 specification and therefore expects its particles will be within the AGR test ranges reflected in Table 5-5 of the EPRI TRISO topical report (Reference 13). As suggested in the SER, Kairos Power will provide justification for any limited discrepancies from the ranges in Table 5-5.

The SER conditions and limitations will be met prior to qualification of the KP-FHR fuel as described in Table 4-1.

### 3.2.3 AGR-3/4

The AGR-3/4 irradiation test was performed to investigate the transport and release of fission products. The data collected from this test is anticipated to be used to inform fuel performance models in support of the source term for a reactor design, but it is not used directly in the KP-FHR fuel qualification methodology. The fuel compacts in the AGR-3/4 contain a set number of designed-to-fail TRISO fuel particles as a known source of fission products. The volatile gaseous and metallic fission products were released from the failed particles and diffused through the fuel compact and its surrounding annular rings of matrix and graphite materials. The gaseous fission products were released to the capsule sweep gas with gas radioactivity measured with a gamma ray spectrometer. The metallic fission products would be retained in the compact matrix or in the annular rings as colder regions of the test capsule. These irradiation tests have been completed and PIE is currently underway.

### 3.2.4 AGR-5/6/7

The AGR-5/6/7 irradiation test was performed to qualify particles in sufficient quantities to demonstrate performance of TRISO fuel particles under normal operation and accident conditions. The irradiation

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portion of the AGR-5/6/7 test campaign was completed in July 2020. A significant number of TRISO fuel particles in this test are irradiated at lower temperatures than AGR-1 and AGR-2.

The AGR-1, AGR-2, and AGR-5/6/7 tests are summarized in Table 3-5. The AGR test conditions are provided in Table 3-6. The AGR-5/6/7 data is not specifically relied upon for qualification of KP-FHR fuel, but represents useful comparative information.

### **3.2.5 Safety Testing**

#### **3.2.5.1 AGR Safety Testing**

Fuel particle performance was assessed in the AGR-1 and AGR-2 safety testing program and is documented in Reference 13. This “safety testing” was performed in dry helium at isothermal temperatures of 1600, 1700, and 1800°C for durations up to 300 hours. The results demonstrated robust performance of the AGR TRISO fuel particles. These test temperatures are significantly higher, and the durations are significantly longer than expected in transients in a KP-FHR. These tests were performed at the Fuel Accident Condition Simulator furnace at INL and the Core Conduction Cooldown Test Facility (CCCTF) at ORNL. Fifteen such tests were performed on AGR-1 fuel compacts and sixteen safety tests (twelve on UCO compacts and four on UO<sub>2</sub> compacts) have been performed on AGR-2 fuel compacts. The Table 3-7 was reproduced from Reference 13 to summarize the furnace safety test results from the AGR program. The TRISO fuel particle failure fractions demonstrate a small increase in the number of failures above 1600°C, which is well above the anticipated fuel temperatures in a KP-FHR.

The test temperatures (1600-1800°C) and time at temperature (up to 300 hours) of the AGR-1 and AGR-2 safety testing program conservatively bound reactivity and thermal hydraulic accidents anticipated in the KP-FHR. This is because the primary challenges to the integrity of the fuel particles are phenomena associated with extended times at elevated temperatures, which allows time for diffusion of fission products through the kernel and coating layers and potential subsequent chemical attack of the SiC layer. The lower temperatures and durations for transients in the KP-FHR will be confirmed as part of the safety analysis report associated with a license application submitted under 10 CFR 50 or 10 CFR 52.

#### **3.2.5.2 Other Safety Testing**

Safety testing was also performed by the Japanese, Russians, and Germans in References 6, 57, 58, and 59.

Reactivity-initiated accidents are events in a reactor that lead to an overpower condition where an increase in fuel temperature can result in fuel element damage or failure and the subsequent release of fission products. Some types of fission power reactors, particularly those that use metallic clad fuel elements, can have postulated reactivity-initiated accidents that may result in severe fuel damage. A summary journal article on tests for TRISO fuel provides background information on TRISO fuel particle performance in simulated RIAs (Reference 59). These types of events can be simulated in a transient test reactor that generates a high neutron flux pulse in the presence fuel test specimens. A few of these types of experiments have been performed on TRISO fuel particles as loose fuel particles or in a carbon matrix-based fuel form, and are summarized in Table 3-8. Japan conducted this type of test in the NSRR (Nuclear Safety Research Reactor) and Russia performed tests in the Hydra (IIN-3M Research Reactor) and the IGR (Impulse Graphite Reactor) facilities (as presented in References 57 and 58). In both cases fresh TRISO fuel particles with UO<sub>2</sub> fuel kernels were tested. The Japanese tests included irradiated fuel although results from these tests are not reported in the literature. The design of the Japanese TRISO fuel particles is similar

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to traditional German TRISO fuel particles reported in Table 2-1 while Russian particles had variations in kernel diameter and coating layer thicknesses.

In these tests, the neutron pulse width had a duration of a few milliseconds leading to nearly adiabatic heating of the fuel particle kernel from an energy deposition of 100 to 2300 J/g-fuel. The IGR fuel test was unique in that the pulse width was long in duration and the energy deposition was very high (>10,000 J/g-fuel). The objective of the IGR test was to determine the energy deposition that would damage the structural integrity of a 60mm diameter fuel pebble. All of these transient tests resulted in very high ramp rates for fuel temperature of >1,000 °C/s, which is much greater than an FHR reactivity transient. In comparison, the rate of temperature change in an FHR is expected to be in the 1 to 10 °C/s range for rapid reactivity insertion events. The rate of temperature change in AGR furnace safety tests was in a lower range of 0.01 to 0.10 °C/s for cylindrical fuel compacts. The rate of temperature increases in the Russian and Japanese transient tests bound conditions in FHR transients by over an order of magnitude. The reported fuel temperatures for a given amount of energy deposition and measured fraction of TRISO fuel particles failures are reported for the most recent set of NSRR tests in Table 3-9.

ORNL performed a modeling study of the Japanese NSRR tests using the fuel performance code BISON (Reference 62). The modeling effort investigated fuel temperatures with special attention on peak kernel temperature and stresses in the SiC layer. General agreement was found between the model and experiment for the prediction of TRISO fuel particle failures. The fuel failures in the experiment correlated with high tensile stresses in the SiC layer and melting of the fuel kernel as predicted by the model. The SiC layer was predicted to be in compression at energy depositions less than 600 J/g- $\text{UO}_2$ . SiC layer failures are not expected when the SiC layer is in compression and this appears to agree with the test results of no failures at lower energy depositions. A second consideration not addressed by the modeling study or the transient reactor tests of TRISO fuel particles is the transient fuel performance of irradiated fuel particles. Irradiated fuel particles will have higher coating layer stresses due to internal gas pressure from fission product gases and CO gas in the fuel kernel and buffer layer porosity. This gas pressure is greater for TRISO fuel particles with  $\text{UO}_2$  fuel kernels since UCO fuel kernels suppress CO gas formation, which accounts for the majority of the gases in irradiated  $\text{UO}_2$ . The gas pressure in a TRISO fuel particles with a UCO kernel would be less than in a fuel particle with  $\text{UO}_2$  kernel at the same burnup and this would result in lower coating layer stresses in transients.

Similar data to that in Table 3-9 is reported for the Russian Hydra and earlier Japanese NSRR tests in Figure 3-1. The peak fuel temperature for these data points is not reported in the references. The combination of all test data suggests an energy deposition of 400 to 600 J/g- $\text{UO}_2$  does not result in damage to the fuel particles. A threshold value appears to occur above 1000 J/g- $\text{UO}_2$ , a large increase in the failure fraction is reported above this energy deposition in all tests. In all cases, the fuel kernels fracture and at higher powers melt and eventually vaporize. The density changes with melting and increased pressure generated from vaporization of  $\text{UO}_2$  are stated as the failure mechanisms.

The German program conducted several furnace safety tests on low enriched uranium (LEU)  $\text{UO}_2$  TRISO fuel particles in 60 mm diameter fuel pebbles as summarized in Table 3-10 (Reference 6). These fuel pebbles were obtained from the AVR reactor and from irradiation tests in test reactors. The tests were conducted at an isothermal temperature of ~1600°C for up to 500 hours with some pebbles more directly simulating accident conditions expected in the HTR Module reactor design. These tests determined that for the German TRISO fuel particle design (Table 3-1) the allowable accident temperature may be greater than 1600°C up to a burnup of 11 %FIMA without additional fuel failures. This report suggested that a

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strict limit of 1600°C would prevent fuel particle failures up to 15 %FIMA for the UO<sub>2</sub> TRISO fuel particle design.

### 3.2.6 Summary

Extensive testing and operating experience in prototype reactors in the UK, Germany, China, Japan, South Korea, Russia, and the U.S. over several decades support the robust nature and lower failure propensity of the TRISO fuel particle. This experience covered a wide range of conditions and is summarized in Table 3-1. Although much of this experience was with UO<sub>2</sub> rather than UCO; this experience is still relevant and applicable because UCO-based fuel is more robust than its UO<sub>2</sub>-based counterpart. This experience includes both steady state operation and transient operation (safety testing).

The German TRISO experience indicated failure fractions consistently in the 10<sup>-5</sup> range for both normal operation and safety testing. For example, German testing of 60 mm diameter spherical fuel elements (including 277,000 particles) showed failure fractions of <1.1x10<sup>-5</sup> at a 95% confidence level (Reference 13). Additionally, in a compilation of German irradiation and safety testing, Kania et al showed five failures out of 287,480 particles, or a failure fraction of <3.7x10<sup>-5</sup> at a 95% confidence level (Reference 13).

The 95% upper confidence bound for the aggregate measured TRISO failure fraction during AGR-1 and AGR-2 irradiations is ≤2.3x10<sup>-5</sup>. This measured performance value is about a factor of ~9 better than historical MHTGR design specifications. Additionally, the aggregate measured SiC failure fraction, defined as the loss of integrity of the SiC layer with at least one remaining intact PyC layer is ≤3.6x10<sup>-5</sup> for AGR-1 and AGR-2, i.e., at the same low level as the TRISO failure fraction (Reference 13).

For transient operation, safety testing was performed by the AGR program, and also by the Germans, Russians, and Japanese. These tests also show that the TRISO particle is very robust under severe transient conditions. These tests were performed at transient conditions beyond those achievable in a KP-FHR.

The fuel qualification limits in Table 3-11 are based on the range of TRISO testing in AGR-2 and provide the KP-FHR envelope for normal full-power operation and AOOs and DBEs. If the fuel qualification limits are exceeded by expected fuel operating envelope, additional testing may be needed as described in Section 3.7.

Future license applications for commercial electric power KP-FHRs will include additional justification (testing or analysis based on an approved methodology) of the applicability of this methodology during rapid reactor transient events for irradiated fuel.

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### 3.3 FUEL PEBBLE PIRT

Kairos Power conducted a design-specific PIRT for the KP-FHR fuel to identify phenomena related to the fission product transport and release figure of merit. As described in the PDC topical report, Kairos Power has selected SARRDL as the figure of merit for the KP-FHR (Reference 3). The purpose of the PIRT is to make sure that the physical, chemical, and irradiation behavior of the fuel is sufficiently understood so that the fuel can be operated safely and modeled adequately.

From 2002 to 2004, a TRISO fuel particle PIRT was performed in support of the NRC review of TRISO fuel development (References 20, 21, 22). This report has been updated by Kairos Power incorporating the knowledge gained from the AGR program experience between 2004 and 2019. The emphasis of the Kairos Power report has also changed from TRISO particles with UO<sub>2</sub> kernels to TRISO particles with UCO kernels. In addition, the scope of the Kairos Power PIRT included the complete fuel pebble, not just the fuel particle. The purpose of the PIRT exercise is to identify areas where additional information is needed to fully understand the behavior of the KP-FHR fuel. The fuel qualification methodology addresses these areas through development, analysis, and testing.

#### 3.3.1 Scope of the PIRT

The focus of the Kairos Power PIRT includes the two major components of the fuel pebble, the TRISO fuel particles and the annular fuel pebble itself as described in Section 2.2. In addition, the service environment is considered in the PIRT as described for the reactor design and molten Flibe coolant in Section 1.1.2.

#### 3.3.2 Scenarios

The specific LBE scenarios that apply to the KP-FHR are determined as part of the implementation of the risk-informed performance-based methodology described in a separate Kairos topical report (Reference 23).

For purposes of this PIRT, a general set of scenarios that are relevant to previous experience with HTGR designs are applied to the KP-FHR fuel PIRT and are considered sufficiently representative to support identification of important phenomena. The relevant scenarios for consideration are listed below and referenced in PIRT summary tables, Tables 3-2 and 3-3.

##### Scenarios Outside of the Reactor

- Manufacturing TRISO Fuel Particles and Annular Fuel Pebbles
  - Manufacturing Process Control and Parameters (MFG-PC&P)
  - Manufacturing Specifications (MFG-Specs)

- Fuel Pebbles in the PHSS

##### Scenarios Inside of the Reactor

- Normal Operation (NO)
- LBEs
  - AOOs
  - DBEs, DBAs, Beyond Design Basis Events (BDBEs), and PHSS Accidents (discussed under reactor accidents for simplicity)

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- Thermal Hydraulic Accidents
- Reactivity Accidents

### 3.3.3 Importance and Knowledge Level Ratings

Phenomena were identified for each component and sub-component of the KP-FHR fuel pebble for each given scenario. Subject matter experts assigned one of three ratings for phenomenon importance to the figure of merit and the level of knowledge. The three ratings are low, medium, and high. The following definitions of these ratings were used in the PIRT.

#### Importance

- High (H): phenomenon has critical influence on evaluation criteria, or improved understanding is critical for making decisions
- Medium (M): phenomenon has moderate influence on evaluation criteria, or improved understanding is important for making decisions
- Low (L): phenomenon has minimal influence on evaluation criteria, or current understanding is adequate for making decisions

#### Knowledge Level

- High (H): knowledge base is adequate for modeling, analysis, or decision making (approximately 70-100% of complete knowledge and understanding)
- Medium (M): knowledge base is incomplete for modeling, analysis, or decision making (30-70% of complete knowledge and understanding)
- Low (L): knowledge base does not exist for modeling, analysis, or decision making (0-30% of complete knowledge and understanding)

### 3.3.4 PIRT Rankings

The expert panel rated phenomenon importance and knowledge level according to the figure of merit. The panel results were averaged, and a combined (importance and knowledge) ranking was assigned to the phenomenon. The rank 1 (H,L) has a high importance and low knowledge level rating. This indicates that more knowledge is needed for the phenomenon. A ranking of 2 (H,M) was defined as a high importance and medium knowledge level, and a ranking of 3 (M,L) was defined as a medium importance and low knowledge level. The phenomena ranked 1, 2, or 3 for TRISO fuel particles are provided in Table 3-2, and the 1, 2, or 3 rankings for pebbles are provided in Table 3-3. Figure 3-2 summarizes the rankings



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for both particle and pebbles. From these results it is concluded that much more is known about the TRISO fuel particle than the KP-FHR fuel pebble on a relative basis.

These rankings are used to identify and prioritize testing, analysis, and code development related to fuel qualification. Knowledge will be improved through technology development and testing programs. PIRT rankings of 1 (H,L), 2 (H,M), and 3 (M,L) are given the highest priority.

#### **3.3.4.1 TRISO Fuel Particle High Priority PIRT Rankings**

The general observations for phenomena rankings are listed by scenario. All primary ranking averages were a rank 2 (H,M) and there were no average rank 1 (H,L) or 3 (M,L) phenomena for TRISO fuel particles.

- Manufacturing – All phenomena with a ranking of 2 (H,M) are associated with the IPyC and SiC layers. This includes the anisotropy in the IPyC layer and bond strength with the SiC layer. For the SiC layer, the identified phenomena are defects and fracture strength.
- NO and AOOs – All of the primary TRISO fuel particle phenomena had a ranking of 2 (H,M) for this scenario, the combination of normal operations and AOOs. The IPyC layer irradiation behavior has the most identified phenomena with inter-layer interactions being important. The diffusion of fission products through the fuel kernel and SiC layer, and chemical corrosion of the SiC layer were also identified phenomena with an average rank of 2 (H,M) for these scenarios.
- PHSS – The potential to crack TRISO fuel particles while in the fuel pebble handling system was considered to have a ranking of 2 (H,M) during normal operations for this system.
- DBEs, DBAs, and BDBEs – There was one phenomenon with a rank of 2 (H,M) for this scenario. This was cracking of the SiC layer.

#### **3.3.4.2 Actions to Address TRISO Fuel Particle High Priority PIRT Rankings**

##### Manufacturing

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#### 3.3.4.5 PIRT Summary

The actions taken as a result of the PIRT for all Rank 1, 2, and 3 phenomena for both the particle and the pebble are summarized in the final column of Tables 3-2 and Table 3-3.

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### 3.4 FUEL SPECIFICATION, MANUFACTURING, AND QUALITY CONTROL THROUGH INSPECTION

The KP-FHR fuel pebble specification describes the pebble design requirements the fuel manufacturer must meet to ensure a level of fuel performance consistent with the KP-FHR design and existing TRISO particle qualification data. The pebble fabrication process will be demonstrated through a manufacturing development program with the fuel specification met through the implementation of a rigorous quality control program. The manufacturing development program will take place in the pebble development laboratory, which is a laboratory scale facility for developing the fabrication and characterization processes for the annular fuel pebble. The developed processes will be transferred to an industrial scale facility for making the commercial KP-FHR core. The manufacturer quality control program will use process controls and product inspections to ensure there is a low fraction of TRISO fuel particle manufacturing defects and to minimize heavy metal contamination. These controls are consistent with national laboratory experience gained in the AGR program.

The KP-FHR fuel particle specification is derived from AGR fuel particle specifications for parameters that are determined to be important to fuel performance. The fuel pebble specifications are also similar to historical German HTGR pebbles for some important parameters. Parameters unique to the annular fuel pebble have nominal values with tolerances in the fuel specification. The fuel specifications will include material composition, dimensions, dimensional tolerances, manufacturing processes, and drawings. The development of these specifications was informed by industry experts with experience in TRISO fuel particle and pebble manufacturing.

The fuel specification, manufacturing, and quality control ensure that the KP-FHR fuel is equivalent to the AGR TRISO fuel particle which is the foundation for the qualification.

#### 3.4.1 TRISO Fuel Particle Specification

The Kairos Power TRISO fuel particle specification is derived from the AGR program fuel specifications where irradiation performance was demonstrated in irradiation tests. Manufacturing the KP-FHR fuel to this fuel specification along with implementation of a quality control program will produce fuel that is equivalent to the fuel used in the AGR irradiation tests. A description and purpose of the TRISO fuel particle and its sub-components are provided in Table 2-2. The fuel particle specification is provided in Table 3-12. The manufacturing defect fractions in the specification are listed in Table 3-13. The TRISO fuel particle manufacture specification related phenomenon in the fuel element PIRT from Table 3-2 will be addressed in the KP manufacturing development program.

#### 3.4.2 TRISO Fuel Particle Manufacturing

The manufacturing process used for the Kairos Power fuel particle will be similar to the AGR program. A general description of the TRISO fuel particle manufacturing process is described in References 5 and 24. A brief description of the process anticipated to be used by Kairos Power to manufacture particles is provided, and key steps are outlined in Figure 3-3 and Figure 3-4.

The manufacturing process begins with the fuel kernel, which is fabricated using a sol-gel process starting from  $U_3O_8$  source material. The sol-gel process involves creating a uranium aqueous broth which is dropped from a vibrating nozzle to form a microsphere. The microsphere or fuel kernel is aged, washed,

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and dried. The kernel is then calcined and reduced with a final sintering step to obtain a high density (> 95% of theoretical).

The fuel kernels are coated in a fluidized bed using a chemical vapor deposition (CVD) process to apply the buffer, IPyC, SiC, and OPyC coating layers. There are four coating process steps one for each of the coating layers. The buffer layer is formed in acetylene gas in an argon carrier gas. The IPyC and OPyC coatings are formed using a mixture of propylene and acetylene gas in an argon carrier gas. In these process steps, the organic compound decomposes through heating and coats the particle with solid carbon. The SiC layer is formed by the CVD process using methyltrichlorosilane gas in a hydrogen carrier gas. All layers are applied in an uninterrupted continuous process in the same coater.

A 2-inch laboratory scale coater was used in the AGR-1 test campaign. An engineering scale 6-inch coater was used in subsequent AGR campaigns. Industrial coaters have ranged in size from 6-inch to 10-inch in diameter. TRISO fuel particles for the KP-FHR will be manufactured using industrial scale fabrication equipment.

### 3.4.3 Annular Pebble Fuel Specification

The fuel pebbles in the KP-FHR have an annular design compared to traditional HTGR pebbles. The annular pebble design has three regions as described in Section 2.2.2 and shown in Figure 2-1. The pebble carbon matrix material is similar to the A3 material used in German HTGR fuel pebbles (Reference 5). Each layer of the fuel pebble serves a specific function as described in Table 3-14. The design specifications for the KP-FHR annular pebble fuel are summarized in Table 3-15.

### 3.4.4 Annular Pebble Fuel Manufacturing

The pebble manufacturing process used for Kairos Power will leverage experience with carbon matrix material from the AGR program experience in addition to information from programs and experts that have developed and fabricated fuel pebbles. A description of fuel pebble manufacturing in open literature discusses the efforts of German and Chinese programs for HTGRs (References 5, 14, and 15). The annular fuel pebble is an improved design as described in Section 2.2.2 with differences from these traditional HTGR pebble designs, however significant similarities exist. The Kairos Power carbon matrix material uses the same key parameters that had historically led to excellent irradiation performance in the German program. The mixture of natural to synthetic graphite is a four to one weight ratio, and the carbonizing step and final annealing are similar. The differences with the German program are in the suppliers of materials and the forming process of the pebble due to the design of the annular fuel pebble. The final product is inspected to demonstrate that it meets the fuel specification.

A flowchart of the anticipated Kairos Power manufacturing process is provided in Figure 3-5. The fuel pebble fabrication process and controls will be demonstrated through a manufacturing development program. This activity will address PIRT phenomenon related to fabrication of the fuel pebble in Table 3-3.

The pebble fuel is primarily composed of a carbon matrix. The carbon matrix is a mixture of natural graphite and synthetic graphite in a four to one weight ratio with a binder material:

- ~64 wt% Natural flake graphite
- ~16 wt% Synthetic graphite
- ~19 wt% Binder material

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- ~1 wt% Curing agent

The fuel pebble will be fabricated in a three-step pressing and molding process. The inner porous lower density carbon core will be formed first. The second step is the forming of the fuel region shell from the overcoated TRISO particles around the inner core. The TRISO particles are overcoated with carbon material and subsequently mixed with the matrix carbon powder in a mixer and then pressed into the mold for the fuel region. The particles are pressed such that there is a minimum inter-spacing between particles in the fuel region, with a nominal packing fraction in the forming process. The third step is the forming of the fuel free carbon outer shell from the same matrix material used for the inner core. The completed pebble may then be machined to dimension. This is followed by two heat treatments that are performed to carbonize and anneal the green (newly formed) pebble. The pebble is heated in a furnace with an argon atmosphere to carbonize the binder, thereby increasing the pebble strength. The carbonization step is followed by a very high temperature heat treatment for an hour to further eliminate impurities and obtain the final material properties.

### 3.4.5 Quality Control and Inspection

A quality control program for fuel manufacturing will be implemented in the fuel manufacturing process with the quality of TRISO fuel particles and pebbles being maintained through inspection demonstrating that the fuel specification is met. Several inspection methods will be used to examine different parameters that are important to irradiation performance. These inspections will take place at different hold points in the manufacturing process; an anticipated list of inspections is provided in Table 3-16.

The UCO kernels, TRISO-coated particles, and fuel pebbles will be fabricated in batches with the batch size being dependent on the manufacturing equipment. Multiple batches will be combined to form lots. Product inspection will be performed using a statistical sampling method.

The purpose of statistical sampling is to verify compliance to the acceptance criteria of all relevant fuel parameters. Specific sampling procedures are put in place for characterization methods using small samples (e.g., kernel diameter, densities, etc.). A list of such properties and adequate sampling procedure will be developed and used by the fuel fabricator. The fuel characterization processes and procedures will be implemented to ensure the fuel meets its specifications.

Due to the importance of the SiC layer, one inspection method of particular importance is performed to determine the SiC defect and heavy metal contamination fraction for fuel pebbles. The heavy metal contamination fraction is comprised of the dispersed uranium fraction (DUF) and the exposed kernel fraction (EKF). Heavy metal contamination is defined as uranium not contained within all three gas-tight layers (IPyC, SiC, or OPyC). The DUF represents uranium dispersed outside of at least one of these layers, while the EKF represents uranium in kernels that are exposed because of broken or cracked coating layers. These two forms of exposed uranium are differentiated because the release fraction of fission products from dispersed uranium is significantly higher than that from exposed kernels. These factors result in the release of fission products to the coolant contributing to the monitored circulating activity in the primary coolant and to the KP-FHR source term.

The deconsolidation leach burn leach (DLBL) method was developed by the AGR program to quantify the DUF, EKF, and SiC defect fractions with a statistical confidence bound. The first acid leach in the process determines heavy metal contamination, which includes a measurement of particles with exposed kernels. The second leach, which follows burn off of the OPyC layer and a segment of the IPyC layer adjacent to

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the SiC defect, determines the fraction of particles with defective SiC layers not associated with exposed kernels.

The DLBL technique will be used to measure the defective TRISO fuel particle fractions during fuel manufacturing. Historically dispersed uranium, exposed kernels, and SiC layer defects have been primary contributors to the fuel element source term in HTGRs, and these defect fractions will be known for KP-FHR pebbles. The DLBL addresses the PIRT issue on the effect of the annular fuel pebble form on TRISO fuel particles in the as manufactured state.



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### 3.5 FUEL OPERATING ENVELOPE

The fuel operating envelope is the bounding set of KP-FHR normal operating and LBE conditions that fuel will be expected to experience. The KP-FHR fuel operating envelope is established in license application safety analysis and is compared to the fuel qualification envelope (Table 3-11) to confirm that expected normal operation and LBE conditions are within the qualification base of the fuel.

Previous fuel qualification and operating experience (Reference 13) have shown that the following parameters are the primary operating conditions that affect fuel performance:

- Power Density** – Power density directly affects the temperature and temperature gradient in the particle, which can impact the fuel particle coating layers driving diffusion and other processes. Kernel migration is associated with temperature gradients as a result of high particles powers. This phenomenon is mitigated by the uranium carbide in the UCO kernel by limiting the formation of CO and CO<sub>2</sub> gas. Additionally, the coolant inlet and outlet temperature in the KP-FHR are relatively low compared to the AGR test data (see Figure 3-7, time averaged temperature). This keeps the fuel temperature in a lower temperature range that prevents kernel migration and reduces the diffusion rate of fission products despite the high particle powers.
- Burnup (Normal Operation)** – Accumulated fissions during normal operation leads to a buildup of fission products primarily in the kernel, but also in the buffer layer for some radionuclides, and into and through the IPyC layer if the layer is compromised. Fission-product mobility is affected by chemical form, which can change with changes in the chemical environment (e.g., oxygen potential in the kernel). Gaseous fission products, as well as carbon monoxide produced by excess oxygen resulting from fissioning of UO<sub>2</sub>, can exert excessive pressure on the dense PyC and SiC coatings. Some metallic fission products (e.g., palladium) and CO can diffuse to and attack the SiC layer, weakening the particle. Note that the KP-FHR will use UCO which mitigates CO effects as long as the burnup does not consume all of the uranium carbide in the kernel.
- Temperature (Normal Operation and Accidents)** – Diffusion and chemical reaction processes along with thermal creep are highly sensitive to fuel particle temperature, depending on the temperature range. In normal operation, the particle temperatures are relatively low, but the duration of exposure is long. If normal operation temperatures are excessive, fission products can diffuse through (e.g., silver) or attack (e.g., palladium) the SiC layer, leading to fission-product release or lessening the capability of the particle to perform acceptably under accident conditions. Temperatures under accident conditions can exceed those for normal operation, although durations are usually much shorter. These higher temperatures can accelerate phenomena that may be present in normal operation (e.g., cesium diffusion).
- Fast Fluence (Normal Operation)** – Interactions with high-energy neutrons ( $E > 0.1$  MeV) during normal operation especially for the fuel particle PyC layers and pebble carbon matrix material can result in irradiation creep, shrinkage, swelling, and changes in important material properties (e.g., strength, density, diffusivity).

The KP-FHR fuel qualification envelope defines the power density, burnup, temperature, and fast fluence that the fuel will be qualified for. The operating envelope is defined for the following set of conditions:

- Normal Operation

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- LBEs
  - AOOs
  - DBEs
  - DBAs
  - BDBEs

### 3.5.1 Normal Operation

The fuel pebble qualification envelope includes reactor normal operation, where normal operation includes reactor startup and shutdown, and operations at low to full power conditions. The test temperature range will be targeted for the equilibrium core in the irradiation test. The range of conditions experienced by the fuel in the KP-FHR cores during normal operation is provided in Table 3-17 alongside the range of conditions at which TRISO fuel particles were tested in the AGR-1 and AGR-2 irradiation tests.

### 3.5.2 LBEs

A detailed set of LBEs (which include AOOs, DBEs, DBAs, and BDBEs) will be determined based on implementation of the KP-FHR Risk-Informed Performance-Based Licensing Basis Development Methodology (Reference 23) and included in the safety analysis reports for license applications. These events are classified depending on expected consequences from scenarios being considered, and frequency of occurrence of these scenarios based on probabilistic risk assessment (PRA). An anticipated fuel operating envelope is compared to the fuel qualification envelope in Table 3-11. This table includes NO, AOOs, DBEs, DBAs, and BDBEs and is provided as an example.

Alternatively, a Maximum Hypothetical Accident may be identified that bounds all potential events and the MHA is evaluated to ensure it is within the fuel qualification envelope.

In either case, the safety analysis reports will confirm that LBEs will remain within the fuel qualification envelope.

### 3.5.3 Fuel Qualification Limits

The qualification limits are based on the irradiation parameters of AGR-2 irradiation tests for power, burnup, temperature and fluence. The AGR-2 irradiation test data sets the qualification limits because the AGR-2 TRISO fuel particle design is nearly identical to the Kairos Power TRISO fuel particle. The values for these irradiation parameters are given in Table 3-11 and originate from the EPRI TRISO topical report (Reference 13). These parameters are described in Section 3.5 as the primary operating conditions that affect fuel performance. A KP-FHR must have an operational performance envelope within these specified limits. A non-power test FHR is shown to operate within this envelope in Figure 3-7. However, the example commercial electric power FHR would operate outside the bounds of this envelope. Additional irradiation test data would be needed to support the operation of the commercial electric power reactor to its full operational envelope. This data would be obtained following the irradiation test methodology described in Section 3.7.

### 3.5.4 Fuel Operating Margin

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There is expected to be significant operating margin to the fuel qualification limits in Table 3-11 during both normal operation and transients for both the non-power test and commercial electric power KP-FHR. For example, normal operating SiC layer temperatures are in the range of 800°C compared to the fuel qualification peak temperature limit of 1360°C.

Similarly, the transients in a KP-FHR are expected to be mild. Consider the example of a limiting reactivity-initiated event, such as the inadvertent withdrawal of a control element, which results in a reactivity insertion that increases reactor power causing an increase in fuel temperature and potential increase of particle failure probability. The inadvertent withdrawal of a control element is considered in this example because control rod ejection is not credible in a KP-FHR due to the low-pressure (~0.2 MPa, ~30 psi) compared to an LWR (~15 MPa, ~2,200 psi) or HTGR (~7 MPa, ~1,000 psi). The driving force to eject a control element is minimal. The inadvertent withdrawal of a control rod is limited by the rate at which the control rod drive mechanism can withdraw the rod. This is limited by design to reduce the rate of reactivity insertion.

The amount of excess reactivity in the core is also limited by the KP-FHR online fueling capability making it possible to change the fissile content in the reactor during operation. As a result, the maximum excess reactivity in a KP-FHR is on the order of 1-2%, in contrast to an LWR where excess reactivity at the beginning of a cycle must be higher (maximum excess reactivity on the order of 7%) to keep the reactor at full power over the 12 to 24 month cycle. The KP-FHR also has several characteristics that limit the impact of reactivity-initiated transients; including strong Doppler feedback (-4 to -5 pcm/°C), short fuel thermal time constant (20ms to 300ms), and long neutron migration length (~25cm) and generation time (~10<sup>-4</sup>s). The energy deposition in the fuel during transients is limited by the transient time with respect to the fuel thermal constant (defined as heat capacity of the fuel divided by the heat removal characteristics). The power increase during transients is impacted by neutron generation time and Doppler feedback. Longer neutron migration length means the reactivity-initiated accident will have slower rate, because neutrons born will have to travel farther (longer) to cause more fission.

These reactivity-initiated transient events will be analyzed as part of safety analysis reports submitted with licensing applications and provide a more definitive understanding of the time at temperature for LBEs. However, the expectation of these future analyses is that conditions will be similar to the those stated here. The analysis of KP-FHR RIAs will be performed using core design and safety analysis codes. The results of this analysis will be input into the KP-BISON fuel performance code. KP-BISON calculates fuel kernel and coating layer temperatures and stresses in the coating layers over the course of the accident and accounts for the irradiation history of the fuel. The fuel temperatures and coating layer stresses will be demonstrated to be within acceptable limits. These limits for temperatures are the melting point of the fuel kernel (2350°C for the carbide phases) and 1600°C for the SiC coating layer, temperature under which it keeps its mechanical integrity. The transient event increases the fuel temperature and gradients resulting in higher coating layer stresses. KP-BISON will calculate coating layer stresses and determine the failure probability of the coating layers. The expectation is for negligible incremental change in the failure probability of TRISO fuel particles given the anticipated temperatures and time duration of KP-FHR RIAs relative to the transient fuel performance observations in furnace safety and transient reactor tests of TRISO fuel particles.

An example preliminary RIA simulation was performed for a reactivity element withdrawal event in a non-power test KP-FHR using the KP-SAM safety analysis computer model. One dollar of reactivity was added to the core by the withdrawal of a reactivity control element without reactor scram resulting in the core

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reaching [ ]. This event occurred over a one- to two-minute timeframe with the SiC reaching a peak temperature of [ ] and an energy deposition of less than 600 J/g-UCO. The temperature ramp rate for the SiC coating layer was approximately [ ] (compared to 1000°C in the Russian and Japanese safety testing) in this example. The peak fuel kernel temperature was several hundred degrees Celsius less than the 2350°C melting temperature. The duration of the RIA was much longer than the fuel thermal time constant (20ms to 300ms) allowing heat to be transferred out of the fuel kernel, hence reducing the peak fuel temperature. This behavior is in contrast to the previously described transient reactor pulse tests by the Japanese and Russians (Section 3.2.5.2), where rapid transients on the order of milliseconds did not allow heat to be conducted out of the fuel kernel. Those tests simulated an event similar to an LWR control rod ejection that is not a credible event in the KP-FHR due to the low system pressure (~0.2 MPa, ~30 psi).

In addition to the above KP-SAM analysis, a KP-BISON simulation was performed to understand the response of the IPyC, SiC, and OPyC layers during a reactivity insertion event for a power KP-FHR. For the KP-BISON analysis, an extreme case of a [ ] reactivity insertion was simulated. This analysis was performed at the maximum burnup possible in the KP-FHR. The results of this analysis indicate that the SiC layer stays in compression throughout the event. Failure of the SiC layer is expected when the tensile stress in the layer reaches its fracture strength. Therefore, it is expected that a negligible increase in TRISO fuel particle failures will be observed in KP-FHR RIAs. This will be confirmed as part of the safety analysis for any KP-FHR.

These simulations demonstrate that a KP-FHR RIA will have a large amount of margin and short timeframe that is well within the AGR program and Russian and Japanese safety test data envelope. As previously described, the AGR safety tests were conducted between 1600°C and 1800°C for up 300 hours and no fuel failures were observed at 1600°C. The SiC layer temperature should not exceed 1600°C in an RIA. In the Russian and Japanese transient reactor pulse tests the fuel particle failures were due to kernel melting or vaporization. The lower bound temperature for the melting point of a UCO kernel is conservatively set at 2350°C and the peak kernel temperature should be less than the melting point (Reference 73). These two criteria define the transient fuel qualification envelope for TRISO fuel particles and are summarized in Table 3-11.

Both simulations described above were unprotected, i.e., did not credit a reactor trip. A safety related positive flux rate trip is part of the KP-FHR design (for both a non-power test and commercial electric power KP-FHR) and would trip the reactor well before significant reactivity was added and result in significantly lower peak temperatures and greater margins to SiC barrier failure. The presence of a safety-related positive flux rate trip is provided as a limitation in Section 4.2.

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### 3.6 FUEL PEBBLE LABORATORY TESTING

The Fuel Pebble Laboratory Test Program addresses high priority items identified in the fuel PIRT (Section 3.3 and Tables 3-2 and 3-3) supporting the qualification of the KP-FHR annular fuel pebble form. This laboratory program includes mechanical, tribology, buoyancy, and material compatibility testing. These tests are primarily mechanical in nature and will be performed on prototypic fuel pebbles with surrogate TRISO particles and appropriately representative specimens of pebble carbon matrix material that do not contain uranium. All fuel pebble laboratory tests are conducted with Kairos Power pebbles that have a carbon matrix material that is very similar to the German A3 carbon matrix as described in Section 3.4.4. The number of test specimens for each test and condition will be determined by a design-of-experiment study to obtain statistically representative values for properties or behaviors from a test. The results from these tests will also inform fuel performance modeling of pebble interactions described in Section 3.8. Test plans were developed for each of the laboratory tests including acceptance criteria and the number of data points required to meet statistically based acceptance criteria.

The Fuel Pebble Laboratory Testing Program will be conducted over a range of conditions based on normal and transient operation and in compliance with the Kairos Power Quality Assurance (QA) program.

#### 3.6.1 Mechanical Tests

Fuel pebbles experience static and dynamic loads during their in-reactor service life, while in the PHSS, and also during shipping and handling. Laboratory mechanical testing will be performed to determine the conditions that result in pebble mechanical failure due to static loads and impacts. Pebbles will be tested at room temperature. Pebble mechanical models will be validated with test data and used to demonstrate that pebbles maintain their structural integrity under different loading scenarios while in service.

A mechanical compression test will be performed by compressing a pebble between two steel plates until the pebble fails. This test will measure force versus displacement and determine pebble crush strength. This test will be performed on annular fuel pebbles at room temperature. Irradiated fuel pebbles will not be tested, because experience from the German fuel program showed an increase in strength of approximately 20% after irradiation. Strength was also observed to increase with temperature and is not a test variable (References 37, 38, 39). The testing of annular fuel pebbles at room temperature results in the lowest value for crush strength.

Historically, the required crush strength of German pebbles (60mm diameter) at room temperature was 18kN with at least 90.0% of fuel spheres in a lot, at a statistical confidence level of 95.5% having this strength. The smaller diameter (40mm) KP-FHR annular fuel pebble crush strength will be lower with a maximum value expected to be approximately half this value [[ ]] due to the smaller pebble size (Reference 40). The measured value of crush strength is expected to be well above load requirements in the core, PHSS, and during shipping and handling. A value based on the current stage of design is [[ ]]. The limiting load is expected to be due to the insertion of the reactivity shutdown element into the pebble bed during a scram. The loads in the reactor and PHSS are currently expected to be well below [[ ]] The damage to fuel particles will be examined in a subset of fractured pebbles using the DLBL technique to determine the fraction of failed fuel particles.

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An impact test consists of dropping a pebble from a pre-defined distance onto a non-yielding surface with the objective of determining the impact load or number of cycles that results in failure. An alternative test method is to accelerate a pebble to a velocity equal to that of pebble being dropped from a set height and impacting a non-yielding surface. The drop distance in tests will be based on heights and impacts that bound those that could occur in the core, PHSS, and during shipping and handling. The test data will be used to determine design values for dynamic loading for different in-service scenarios to prevent pebble fracture.

The German fuel pebble program dropped 60mm diameter pebbles from a height of 4m onto a bed of pebbles (to simulate fuel insertion during operations) and onto a steel plate from 2m. No specification was identified for impacts on steel plates. The specification requirement was that each fuel pebble and at least 99.5% of fuel spheres in a lot shall at a statistical confidence level of 95.5%, survive 50 drops (Reference 38). In practice with pressed fuel pebbles a sample of pebbles from production lots of German GO2 pebbles for the AVR survived well above 100 drops prior to failure (Reference 5). In practice irradiated fuel pebbles were dropped into the German AVR and THTR reactors multiple times over their lifetime.

The pebbles in the KP-FHR design are not dropped into the core (they are neutrally buoyant in Flibe coolant and are injected at the bottom of the core). The bounding maximum drop height possible in the KP-FHR in the core, the PHSS, and during shipping and handling will be determined. Pebbles will be dropped from this height at room temperature onto a steel plate.

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The following properties or behaviors will be investigated and measured as part of the mechanical testing.

- Pebble force versus displacement under compressive loading and compressive load resulting in failure in a compression test
- Pebble maximum dynamic load or number of cycles in impact tests

### 3.6.2 Tribology

Tribology testing will be performed to quantify coefficients of friction and wear rates during pebble-to-pebble, pebble-to-graphite, and pebble-to-stainless steel contact. These measurements will be made in Flibe and in argon gas. Tribology testing will provide measurement of wear rates that will then be used in a bounding calculation to determine fuel pebble surface wear. Wear rates are used to validate the adequacy of the pebble outer fuel-free zone thickness and to estimate carbon (from pebbles) and graphite (from reflector surfaces) dust generation.

Tribology testing will use the pin-on-disk method, which involves sliding a test carbon matrix pebble against a rotating graphite, carbon matrix, or 316H stainless steel disc while applying a force on the pebble normal to the contact surface. The tribology testing will be informed by the corresponding American Society for Testing and Materials (ASTM) Standard G99 (Reference 75). The normal force will be based on

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the maximum forces of pebble-to-pebble, pebble-to-structural graphite, and pebble-to-stainless steel expected during service. [[

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The contact surface will be immersed in argon gas or in molten Flibe with a controlled argon atmosphere representative of KP-FHR. Test temperatures with Flibe will envelope KP-FHR core operating temperatures [[ ]], and will envelop PHSS operating temperatures [[ ]] with argon. Test temperatures are limited to [[ ]] as this is the upper bound for the operating range of the KP-FHR. While accident conditions may result in higher temperatures, wear accumulation is a long-term phenomenon and would not be significantly affected by short term accident conditions.

The coefficients of friction are directly derived from measuring the tangential force on a sliding carbon matrix pebble with a load cell during the test. The wear volume is measured by microscopy inspection of the wear scar after the test. A specific wear rate (wear volume per unit force and per unit sliding distance) is derived by dividing the wear volume by the normal load and the sliding distance. The effect of wear on the pebble diameter and the amount of generated dust per wear test can be estimated from the wear volume.

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A preliminary calculation using this methodology and based on wear rates measured in molten FLiNaK and argon and using ET-10 graphite as a surrogate fuel pebble materials results in a value of [[ ]] for wear radius, which is less than the 1.5 mm fuel-free outer shell thickness, and therefore would meet

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the acceptance criterion. Discrete Element Models (DEM) are used to simulate the flow of pebbles through the pebble bed and will be used to confirm that the pebble surface wear is significantly less than the value determined through the bounding calculation. Because the DEM is used to confirm the conservative licensing basis calculation of wear, validation of DEM for this application is not provided. However, References 68, 69, 70 provide examples of the use of this methodology.

Friction and wear properties will be measured in the indicated conditions as part of the tribology test program:

- Coefficients of friction for pebble-pebble, pebble-graphite (structural), and pebble-metallic-structure (316H) contact, in Flibe and argon from room temperature to 700°C depending on the test environment.
- Wear rate for pebble-pebble, pebble-graphite (structural), and pebble-metallic-structure (316H) over a range of contact forces, in Flibe or argon from room temperature to 700°C depending on the test environment.

### 3.6.3 Buoyancy and Molten Salt Infiltration

Buoyancy testing will examine the ability of the annular pebble to maintain net positive buoyancy under normal and LBE conditions and investigate salt infiltration into the pebble.

These tests are performed using Kairos Power pebble carbon matrix material that is very similar to the German A3 material having a pore size of one micrometer and [[

]] and at pressures up to [[ ]]. The fuel pebble manufacturing specification defines the pebble density necessary to maintain net positive buoyancy over its service life. The fuel pebble density is measured in the manufacturing inspection process to ensure a maximum density is not exceeded. The thermo-physical properties of density changing due to thermal expansion with temperature will be measured for the pebble carbon matrix and Flibe salt. This data and its uncertainties along with information on the densification of the pebble carbon matrix with irradiation will be used to analytically demonstrate that annular pebbles have a net positive buoyancy in the Flibe coolant.

The acceptance criterion is that measurements of pebble density and Flibe density made over a range from [[ ]]] are assessed analytically to ensure that the pebble remains buoyant over this range of temperatures.

The infiltration of Flibe into the fuel pebble will be investigated at three temperatures [[ ]]] using specimens of pebble carbon matrix material. Testing will be conducted at atmospheric pressure and pressures up to [[ ]]]. The highest pressure in the core is ~200kPa at the coolant inlet during full-power operation. The weight change of the specimen will be measured and specimens will be sectioned and examined to characterize the infiltration of salt.

The infiltration of Flibe into the pebble carbon matrix has been investigated by Chinese researchers for the Thorium Molten Salt Reactor - Solid Fuel (TMSR-SF), an FHR project (Reference 63). In these experiments, carbon matrix material was held at temperature and pressures between 400 kPa and 1000 kPa for 20 hours at 650°C in Flibe. The threshold pressure for infiltration was determined to be 600 kPa, which is well above expected pressures in an FHR (~200 kPa). This value is also in agreement with the 570



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kPa pressure predicted by the Washburn equation (Reference 63) for the infiltration of liquids into porosity with a measured pore size of one micrometer.

ORNL and Chinese researchers have also conducted similar experiments with structural graphite (References 64, 65, 66). The pore size in many grades of structural graphite is larger than in the pebble carbon matrix tested by the Chinese. These experiments with structural graphite show infiltration at lower pressures in comparison to the Chinese tests with pebble graphite. This outcome was also predicted by the Washburn equation, namely that larger pore sizes allow for infiltration at lower pressures.

Irradiation of carbon matrix results in a decrease in pore size with graphite densification (Reference 72). Therefore, the above results that were not performed under irradiation are conservative for the phenomenon of molten salt infiltration. Carbon matrix densification decreases pebble buoyancy with irradiation. The increase in density does not result in the loss of net positive buoyancy. The maximum fluence of a fuel pebble in a non-power test and commercial electric power reactor is also less than the turnaround fluence (the value at which densification reverses) for German A3 materials (Reference 72). The irradiation behavior of graphitic materials is shown to result in a reduction in pore size and the closure of cracks prior to the turnaround fluence.

The penetration of molten salt into graphite was also studied in the Molten Salt Reactor Experiment (MSRE) program (Reference 74). It was determined that fluoride fuel salts have non-wetting characteristics when in contact with graphite. The irradiation and high temperature environment of the MSRE did not alter these non-wetting characteristics. Compositional difference in salts and presence of fission products also did not change the non-wetting behavior. It was determined that the controlling factor for salt penetration into porosity was pressure, where the behavior of salt penetration or infiltration into porosity can be predicted using the Washburn equation.

Both the Chinese and ORNL experiments indicate that Flibe infiltration should not occur into the KP-FHR annular fuel pebbles. Additionally, the non-wetting behavior of fluoride salts in contact with graphite was shown to not be influenced by irradiation or salt composition and impurities in the MSRE program. In summary, the pebble is (1) designed to maintain buoyancy during normal operating conditions and postulated events, (2) inspected during manufacturing to make sure the as-manufactured pebbles are within limits, and (3) tested as described above to confirm that buoyancy will not be lost during operation. For these reasons, gross loss of pebble buoyancy is considered to be a non-credible event.

Future license applications for commercial electric power KP-FHRs will include additional justification (testing or analysis based on an approved methodology) that Flibe does not adversely impact irradiated fuel pebble buoyancy.

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The following properties or behaviors will be investigated and measured as part of the buoyancy test program:

- Demonstrate net positive pebble buoyancy over a range of temperatures from [[ ]]

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- Determine the amount of Flibe infiltration into pebble carbon matrix material over the temperature range of [[ ]] and pressure range of atmospheric to [[ ]]

#### 3.6.4 Material Compatibility

Material compatibility testing examines the interaction of the pebble carbon matrix with Flibe and air. The Flibe tests will determine if the surface of the pebble degrades in the presence of Flibe as a function of Flibe chemistry and temperature. The Flibe chemistry will either be in a reducing clean salt or salt with an oxidizing chemistry that will contain higher levels of impurity metals. The tests will be conducted using coupons of pebble carbon matrix at the temperatures of [[ ]], which bounds the range of fuel pebble surface temperature during normal operation. The duration of the test will be up to [[ ]] hours, which is the time duration of a pebble passing through the non-power test KP-FHR equilibrium core. In Section 3.6.3, molten salt infiltration testing is performed at [[ ]] hours. This experiment determines the molten salt infiltration behavior and also will be used to investigate material compatibility at accident conditions where temperatures are high and the time duration is short. Material interaction in these experiments will be determined by specimen weight or density measurements before and after testing to determine the change in mass. This activity also includes specimen cross-sectioning and microscopy to examine materials for indications of interaction and to determine possible mechanisms.

The interaction of graphitic carbon material with Flibe has been demonstrated to be minimal through laboratory tests and experience with the MSRE. The MSRE was operated with liquid Flibe fuel that was in contact with graphite moderator elements in the core. The Flibe contained uranium and fission products in a high radiation and temperature (650°C) environment with minimal interaction being observed between the materials after three years of operation (Reference 53). Recent short duration tests at the University of Wisconsin-Madison (UW) also show minimal surface interaction of Flibe with IG-110 (graphitic carbon) after a 12-hour test at 700°C (Reference 67). The observed chemical interaction in this experiment was a minimum amount of fluorination at reactive surface sites on the graphite. This behavior has some implications towards tritium pickup on graphite surfaces. A bulk degradation of graphite was however not observed and is not expected based on thermodynamics.

The testing will not investigate the interaction of Flibe with TRISO fuel particles based on the current information available from previous testing. A lack of Flibe infiltration was observed into A3 carbon matrix material in Chinese research for the TMSR-SF preventing the interaction between Flibe and TRISO fuel particles as discussed in Section 3.6.3. The interaction of TRISO fuel particles with Flibe salt has also been investigated at the University of Wisconsin-Madison (UW) with laboratory-based corrosion experiments on surrogate TRISO particles, while Massachusetts Institute of Technology (MIT) conducted identical corrosion tests in the MITR-II reactor (References 46, 47, and 48). The TRISO particles were provided by ORNL that contained a surrogate kernel composed of ZrO<sub>2</sub>.

The UW laboratory test was conducted at 700°C for 1000 hours and the MIT irradiation test was conducted for 300 hours at 700°C, with both tests being performed in Flibe. Additionally, to understand the effect of Flibe during irradiation, surrogate TRISO particles were also irradiated in helium gas for 2200 hours at MIT at 1000°C. The tests were conducted in graphite crucibles, that were degassed/purified in an H<sub>2</sub> furnace prior to loading. The Flibe used was a purified enriched Flibe with its transition metal impurities, as determined by Neutron Activation analysis. The TRISO particles tested at UW in static Flibe

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looked pristine when removed and sectioned for microscopy. No cracks or signs of degradation of any layers were observed.

In the MIT tests, TRISO particles irradiated in Flibe, the researchers observed gross cracking of OPyC layer prior to mounting and polishing. The OPyC cracking was associated with irradiation effects and the freeze/thaw cycles of the Flibe. The FHR-IRP testing showed that no degradation of the SiC layer from exposure to purified Flibe both in and out of reactor. However, the irradiation did prove that the irradiation embrittles the outer layers of the TRISO particles making them susceptible to cracking during either sectioning/grinding, or during repeated freeze/thaw cycles of Flibe. Because the pebble outer fuel-free zone prevents Flibe ingress, fuel particles are not affected by freeze/thaw cycles that occur when pebbles are cooled, examined, and temporarily stored in the pebble handling and storage system.

This program will also investigate the reaction of pebble carbon matrix material with air (oxygen) at an intermediate temperature range associated with accident conditions in the PHSS and reactor vessel. The PHSS and reactor vessel cover gas system will be designed to limit temperatures to be within this intermediate range in the case of air ingress event. The reaction of graphite materials with air has been well studied at high temperatures, however less data is available in the intermediate temperature range (200 to 800°C) and specific to Kairos Power produced pebbles (Reference 49). Therefore, tests will be performed on carbon matrix material in air at relevant temperatures up to 800°C in a furnace. The pebble carbon matrix reacts with oxygen in air at elevated temperatures and generates CO and CO<sub>2</sub> gas, and a loss of specimen mass will be measured to determine the reaction rate. The mass of dust produced from oxidation experiments will also be measured. The experimental data will be used to either validate an existing correlation for carbon matrix oxidation or develop a new correlation as appropriate. This correlation will be used to evaluate the behavior of the fuel pebble in an air ingress event as needed to support safety analysis.

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The following properties or behaviors will be investigated and measured as part of the material compatibility test program:

- Material interaction and degradation of the pebble carbon matrix when in contact with Flibe at reactor operating temperatures.
- Reaction rates between the pebble carbon matrix and air at temperatures relevant to the PHSS.

An overview of the Laboratory Test Program is provided in Figure 3-6. A summary of the acceptance criteria for the tests in the Laboratory Program is provided in Table 3-18.

The laboratory test program is focused on confirming that the pebble retains its integrity during operation in a KP-FHR. Results from this program will be used to inform the fuel performance modeling in Reference 19, if necessary.

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### 3.7 FUEL IRRADIATION TESTING

If the KP-FHR operates outside the range tested in the AGR-2 irradiation (Figure 3-7), a fuel pebble irradiation test will be conducted to expand the qualification of the fuel particle so that it envelopes conditions associated with equilibrium full-power core operation. If the KP-FHR operates within the conditions tested in AGR-2, then additional irradiation testing is not needed.

This section describes an irradiation test for demonstrating qualification of the KP-FHR fuel particles beyond AGR-2 qualification bounds. This section also describes the statistical methods used to translate the experimental results into a failure fraction and the PIE that is performed. The Fuel Irradiation Test Program will be performed in compliance with the Kairos Power QA program.

#### 3.7.1 Fuel Pebble Irradiation Test

This method of KP-FHR fuel qualification is based on fuel pebble irradiation in a non KP-FHR test facility within a gas environment. During this test, the fission gas release rate to birth rate ratio will be measured for noble gas isotopes released into a purge gas. The data will be used to determine the fuel particle failure fraction during the irradiation test with the objective of demonstrating equivalence with existing AGR program data for TRISO particles in non-pebble based compact forms. This test will address both the PIRT issue related to the impact of fuel form on the TRISO particle and the qualification of the pebble for equilibrium core conditions.

In a pebble bed reactor, the irradiation history of fuel pebbles is stochastic because fuel pebbles can traverse many different paths through the core. Instead of replicating the many possible irradiation histories that might exist in the KP-FHR, this irradiation test method will target bounding conditions for power, burnup, temperature, and fast fluence that are most likely to challenge fuel pebble and TRISO fuel particle reliability with a potential for increasing fission product release. Therefore, the irradiation test will focus on conditions of the KP-FHR equilibrium core as bounding irradiation conditions defined in the fuel qualification envelope discussed in Section 3.5 and listed in Table 3-11. The combination of position in the test facility and the test capsule design will be used to obtain the desired conditions. Fuel pebbles at lower and higher axial positions in the test facility could observe a 30-40% reduction in burnup and fluence from targeted conditions providing a range in test conditions.

The design of the irradiation test capsule is summarized in Table 3-19. Data will be collected from in-test instrumentation that will provide real-time feedback on the test's progress. The objective is to obtain a statistically significant number of fuel particles at the targeted irradiation test conditions. The test is anticipated to include from six to sixteen pebbles depending on the test facility and capsule design.

The test capsule will locate individual pebbles inside a graphite holder structure, and use a combination of a gas gap and controllable sweep gas mixture to maintain the graphite structure temperature within a desired range. The test capsule will be instrumented with thermocouples to provide data on the test temperature and maintain temperature control. Flux monitors are included in the test to measure the neutron fluence. These data will be used to validate neutronic calculations of the irradiation.

The sweep gas in the fuel pebble capsule will be monitored continuously with radiation detection systems to measure gross activity. A steady increase in gross activity is associated with increased fission product release due to increased test temperature, while a significant step increase is due to fuel particle failure. Gamma spectrometry will be performed on the sweep gas to determine the release rate to birth rate ratio

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for gaseous radioactive fission products released from the test articles. The data from online instruments will be recorded by a data acquisition system in an electronic format.

Previous irradiation test data has been collected using capsules with TRISO fuel particles in different fuel forms in an inert gas environment. The irradiation test in a non KP-FHR test facility will use a similar method to allow direct comparison with previous results, most notably AGR results. The effect of Flibe on the pebble form is addressed in separate tests, which are designed to assure that the pebble is not compromised by Flibe, although these tests are not done under irradiation. These tests are described in Sections 3.6.3 (Buoyancy and Molten Salt Infiltration) and 3.6.4 (material compatibility). Since the interaction under irradiation appears to be minimal, the chemical interaction between the Flibe coolant and pebble carbon matrix can be assessed in separate laboratory tests. The integral fuel performance of TRISO fuel particles in annular fuel pebbles in an irradiation and Flibe environment will be confirmed in the KP-FHR fuel surveillance program described in Section 3.9.

### 3.7.2 Post-Irradiation Examination of Fuel Pebbles From non KP-FHR Test Facility

PIE will be performed on the fuel pebbles removed from the non KP-FHR test facility. The PIE will involve NDE and destructive examinations (DE) to demonstrate the irradiation behavior of the fuel pebble. NDE includes visual examinations, dimensional measurements, and gamma spectroscopy to identify gross external damage and burnup. These examinations will provide data on the irradiation performance of the fuel pebble and provide validation data on fuel burnup.

Destructive examinations (DE) will be performed on a limited set of test articles after NDE. The destructive examination will provide a final confirmation on fuel particle failure fractions by performing DLBL tests (Section 3.4.5) on the fuel pebbles. This measurement technique is used in the AGR program and has historically been used in HTGR fuel pebble irradiation test programs to quantify the fuel particle failure fraction in a test with a statistically significant number of pebbles. DLBL allows independent quantification of both TRISO and SiC failures that are experimentally known to exhibit the same low level of failure (see Section 3.2.6). Data from these DLBL tests will confirm the low fuel failure fraction measured by fission gas release during the irradiation test.

The number of pebbles undergoing DE could range from [[ ]] depending on the goal failure fraction the program is attempting to achieve with a 95% one-sided upper confidence bound.

If the irradiation testing and associated PIE are completed with acceptable results, the KP-FHR fuel is qualified for startup and equilibrium operation.

For this irradiation test method, the destructive examination portions of the PIE program are confirmatory in nature and not required to be completed prior to transitioning to equilibrium operation in the KP-FHR because sufficient information on failure fraction is obtained from NDE pebble irradiation testing.

### 3.7.3 Irradiation Test Acceptance Criteria

[[ ]] . The failure fraction is determined using the statistical methods described in Section 3.7.4. For both tests the acceptance criteria is that the failure fraction is within the allowable failure fraction limit of [[ ]]

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The Section 3.2.6 of this topical report discusses the TRISO fuel particle failure fractions measured in irradiation tests of German LEU UO<sub>2</sub> TRISO fuel particles in pebbles and US AGR Program LEU UCO TRISO fuel particles in cylindrical compacts. The German TRISO fuel program set the standard for irradiation performance and the AGR program demonstrated the ability to replicate this performance in the AGR-1 and AGR-2 irradiation tests. The measured failure fractions from irradiation in these programs were in the range of 1.1 to 3.7x10<sup>-5</sup> (for the 95% upper confidence bounds). The EPRI TRISO fuel performance topical report discusses in-service fuel particle failure fractions in HTGRs (Reference 13). An acceptable range of fuel failure fractions is 1.6 to 2.0x10<sup>-4</sup> in the example reactor cases for normal operations. Based on all of these values Kairos Power applies an acceptance criteria for the irradiation testing at a fuel failure fraction of [ ] as shown in Table 4-1. TRISO fuel that meets this value is considered to be of good quality based on the historic irradiation performance of TRISO fuel and its application in HTGRs. If the irradiation test fuel failure fraction is less than or equal to [ ] then it is judged that the fuel is consistent with past acceptable TRISO fuel performance and the fuel is qualified.

An additional consideration is the use of Flibe coolant in the KP-FHR in comparison to HTGRs. The Flibe coolant retains fission products providing an added barrier between radionuclides from the fuel and the environment in the KP-FHR that is not present in HTGRs. The acceptance criteria is further justified by the KP-FHR Flibe coolant enabling a higher level of fuel particle failures to be tolerated in comparison to a HTGR.

### 3.7.4 Statistical Methods

The following statistical methods are used to determine the number of pebbles needed in an irradiation test or PIE program.

This statistical problem involves a large population with an inherent number of particle defects from manufacturing and the potential for failure during the service lifetime due to damaging mechanical and thermo-chemical phenomena. The KP-FHR core contains thousands of pebbles and each fuel pebble contains approximately 16,000 fuel particles. The sample size in an irradiation test is also large (between ten thousand and one million), failure probabilities are low (less than 1x10<sup>-3</sup>), and a confidence level of 95% is desired for a population. Kairos Power uses a one-sided upper confidence bounds (UCB) for binomial distribution probability (Reference 25, 26). The objective is to identify a conservative statistical method for application in design of an irradiation test. In this case, a test population of fuel particles will statistically represent the greater population of fuel particles present in the KP-FHR core with a 95% UCB.

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This method will be used in the design of the fuel pebble irradiation test operating at limiting irradiation conditions of the KP-FHR equilibrium core. The test will include a statistically significant number of TRISO fuel particles to demonstrate a target failure fraction with a 95% one-sided upper confidence bound. In Table 3-20, the [[ ]] method is used to calculate the number of TRISO fuel particles needed in a test to demonstrate a target failure fraction when zero failures are observed as an example application.

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### 3.8 FUEL PERFORMANCE MODELING

Fuel performance models are used to analyze the response of the fuel during normal operation and LBEs to ensure that the fuel is operating within its design. In addition, fuel performance modeling complements the fuel qualification program with models being used to inform the design of fuel qualification tests. In turn, data generated from the test programs is used to validate and improve the fuel performance models. Fuel performance modeling involves two component level models; a fuel performance model of the TRISO fuel particle and a thermo-mechanical model of the fuel pebble. Each of these models has a specific application in the reactor design and licensing process.

KP-BISON is used to model and simulate the behavior of TRISO particles and fuel pebbles in normal operating and LBE conditions. Specifically, it will calculate the failure probability of the coating layers and the release of fission products, under irradiation or during LBEs.

Fuel performance supports and informs the fuel qualification methodology and will be used to assess future changes to the fuel, but it is not a specific part of the initial qualification of the fuel.

#### 3.8.1 TRISO Fuel Particle Performance Model

KP-BISON is based on BISON, a next-generation nuclear fuel modeling capability developed by INL. The code is applicable to both steady and transient fuel behavior and can be used to analyze either 1D spherical, 1.5D axisymmetric, 2D axisymmetric, or 3D geometries. BISON has been applied to a variety of fuel forms including TRISO-coated particle fuel. The use of KP-BISON for KP-FHR fuel performance is documented in the Kairos Power Fuel Performance Methodology Topical Report (Reference 19). The validation of the KP-BISON fuel performance model with AGR-1 and AGR-2 irradiation test and PIE data (Sections 3.2.2 and 3.2.5) is used in part to address PIRT phenomena in Table 3-2 related to the irradiation performance of TRISO fuel particles. The validated fuel performance is used for core design and source term analysis.

Models included in KP-BISON describe physical phenomena occurring during irradiation of TRISO fuel, such as temperature- and burnup-dependent thermal properties, fission product swelling, irradiation-induced dimensional changes and creep, or fission gas generation and release. The code is a modeling and simulation tool used to calculate the irradiation performance of the constituent parts of a TRISO-coated fuel particle (kernel, buffer, IPyC, SiC, and OPyC coating layers) and spherical fuel pebble in which TRISO particles are embedded.

#### 3.8.2 Pebble Fuel Discrete Element Model

The fuel pebbles in the KP-FHR core are buoyant and move through the core as a random bed of pebbles. Movement of the pebbles inside the reactor core is modeled and analyzed by the Discrete Element Method (DEM). DEM calculates pebble locations, velocities, and spatial porosities. These parameters are used for the reactor physics and depletion calculations. DEM also is used for the explicit computational fluid dynamics simulations to calculate core pressure drop and convective heat transfer in the randomly packed bed environment. DEM relies on property data related to the strength, tribology and wear behavior of the fuel pebbles (Section 3.6). The tribology and related properties are friction and wear coefficients for pebble-to-pebble contact and pebble contact with structural materials in Flibe and gas



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environments. Tribology testing will provide input on the pebble wear rate and behavior for these applicable surface contacts and environments for use in DEM models.

### **3.8.1 Pebble Fuel Finite Element Method Model**

The thermo-mechanical behavior of the fuel pebbles is modeled using a Finite Element Method (FEM) model. The FEM will be used to evaluate pebble stresses while in the reactor and PHSS. The fuel pebbles have a design limit for the maximum load and impact conditions. The purpose of this model is to demonstrate that static and dynamic loads on pebbles during their service life do not lead to unacceptable pebble damage or fracture. The maximum static and dynamic loads for fuel pebbles are determined in laboratory compression and impact tests described in Section 3.6. These tests will provide data that supports both the development and validation of pebble thermo-mechanical FEM models.

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### 3.9 FUEL SURVEILLANCE

The KP-FHR fuel surveillance program monitors fuel performance during KP-FHR operations. It is an ongoing program to ensure the fuel continues to perform as designed. The surveillance program involves online monitoring of fission product activity in the cover gas and coolant, inspection of pebbles in the PHSS during operation, and selected destructive examination of pebbles for the initial KP-FHR core.

The first part of fuel surveillance is the monitoring for radiation from fission products in the cover gas and molten salt coolant, which is an indirect measurement of fuel particle failures.

The second part of fuel surveillance is NDE in the PHSS, examining pebbles for damage and performing gamma spectroscopy to determine fuel burnup. All pebbles are subject to this examination as they exit the core. The fuel pebbles are expected to pass through the core multiple times during their lifetime. These transit times are dependent on core design. Core design and analysis for the KP-FHR are addressed in a separate report.

The third part of fuel surveillance involves destructive examination of selected pebbles in the initial KP-FHR to confirm fuel performance observations from the fuel surveillance program and provide information for fuel performance models.

#### 3.9.1 Monitoring Cover Gas and Coolant Activity

The first and most rapid indication of a fuel performance issue is by radiation detection (circulating activity) in the cover gas system and Flibe coolant as described in Section 2.2.3. The design of the KP-FHR will include monitoring of radiation levels in the cover gas and Flibe coolant during operation to observe fuel performance trends and to meet the requirement of remaining within the SARRDL. If significant changes in radiation levels are observed in an abrupt occurrence or short period of time that is not consistent with expectations, this will require investigation to determine causes. These types of occurrences would first be detected by monitoring fission products in the cover gas and coolant.

#### 3.9.2 Fuel Pebble Non-destructive Examinations

Pebbles are non-destructively examined after leaving the core using an inspection system in the PHSS. The function of the inspection system is to maintain the health of the reactor by preventing the re-introduction of pebbles that are damaged back into the core. The purpose of this action is to prevent the release of fission products.

#### ***NDE***

- Gamma Spectroscopy – determine pebble burnup
- Inspection – identify external damage

The first measure of fuel performance as previously described is performed by monitoring fission product activity in the cover gas and Flibe coolant as an indication of fuel failure. An unexpected increasing trend

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in the radiation level could indicate an issue with fuel pebbles and the inspection system would be used to remove pebbles that could be contributing to the event.

After leaving the core, fuel pebbles are examined by gamma spectrometry in the PHSS to determine the burnup through the measurement of gamma ray activity from signature fission products. Pebbles approaching or at a burnup limit will not be returned to the core and instead be sent to storage.

The PHSS inspection system also examines fuel pebbles for gross damage such as wear, cracking, missing surfaces, etc. Pebbles that show indications of wear, cracking, or missing surfaces will be removed from service. There are two criteria for removal of pebbles due to pebble damage.

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If fission product activity is within the expected range, fuel pebbles are assumed to contain intact TRISO fuel particles within the bounds of the expected failure fraction as determined from the manufacturing specification and fuel performance analysis. Independent of the given acceptance criteria, this observation would indicate that any observed features in the PHSS under this condition might be considered acceptable since they would not appear to correlate with fuel particle failure.

### 3.9.3 Fuel Pebble Destructive Examinations

A set of fuel pebbles from the initial non-power test and the initial commercial electric power KP-FHR core will have destructive PIE performed to confirm fuel performance observations from the fuel surveillance program. This information will confirm the integral performance of the fuel pebble under irradiation while in the high temperature molten salt environment at prototypic KP-FHR operating conditions.

Destructive examinations (DE) will be performed for the initial non-power test and the initial commercial electric power KP-FHR on a limited set of test articles after NDE. The destructive examination will provide a final confirmation on fuel particle failure fractions by performing DLBL tests (Section 3.4.5) on the fuel pebbles. This measurement technique is used in the AGR program and has historically been used in HTGR fuel pebble irradiation test programs to quantify the fuel particle failure fraction in a test with a statistically significant number of pebbles. For example, the acceptance criteria of [[ ]] could be demonstrated with two fuel pebbles if no TRISO particle failures are observed as shown in Table 3-20. A direct comparison to the AGR-1 and AGR-2 irradiation tests would require [[ ]] pebbles to be examined to demonstrate a failure fraction of  $\sim 2 \times 10^{-5}$  if no failures occur in the sample population. The PIE data will confirm the expected low fuel failure from monitoring cover gas space and Flibe coolant for radiation from fission products during KP-FHR operation.

The destructive examination will also include a detailed investigation of pebble wear and Flibe infiltration into the pebble outer fuel-free zone confirming the integral performance of the pebble as observed by the pebble monitoring in the PHSS.

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### 3.10 SUMMARY OF FUEL QUALIFICATION METHODOLOGY

The fuel qualification methodology provides reasonable assurance that the KP-FHR annular fuel pebble design can operate safely in a KP-FHR. This reasonable assurance is based on the following:

- Over thirty years of operating experience and testing of TRISO fuel including extensive testing of TRISO fuel particles in AGR-1 and AGR-2, including for both steady state and transient conditions.
- Successful completion of a KP-FHR fuel element PIRT and implementation of associated actions to further the understanding of the annular fuel pebble and TRISO fuel particles.
- Manufacturing and inspection of the KP-FHR fuel to a specification that ensures the fuel is equivalent in performance to the fuel tested in AGR-1 and AGR-2, and meets the conditions in the TRISO topical report SER (Reference 13).
- Operation within a set of defined fuel qualification limits which ensure that the fuel remains within its qualification envelope during both normal operation and licensing basis events.
- Successful completion of the Laboratory Testing Program which ensures that the pebble will retain its design form and function during normal operation and LBE conditions.
- If fuel pebbles will operate outside of the AGR-2 fuel performance envelope, then irradiation testing will be performed in a non-KP-FHR test facility.
- Confirmation that the pebble form does not have an adverse impact on the fuel particles through a fuel surveillance program in the KP-FHR.
- The ability to examine fuel pebbles as they exit and re-enter the core over their expected lifetime, including the ability to remove them if necessary for disposal or PIE.

In addition to these elements of the qualification methodology, the following KP-FHR design features provide additional assurance that this fuel can be used safely in a KP-FHR:

- The KP-FHR has substantial operating margin to fuel qualification envelope temperature during normal operation which is based on the AGR-2 irradiation tests (normal operation at particle temperatures < [[ ]]] vs. 1360°C maximum temperature in AGR-2 irradiation test).
- During postulated events, significant margin also exists with a conservatively calculated peak SiC layer temperature of [[ ]] being well below the transient limit of 1600°C as discussed in Section 3.5.3. In addition, a KP-BISON analysis of the TRISO particle coating layers during an extreme reactivity transient demonstrates that the SiC layer remains in compression throughout the event.
- During startup and initial operation, there is a large margin to the radiological source term limit. The radiological source term assumes a maximum condition for the accumulation of radionuclides over the lifetime of the reactor, which is not the condition during early operations. One of the key drivers for the KP-FHR source term is actinide impurities in the

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Flibe, which at the beginning of life during startup and initial operation are expected to be minimal.

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## 4 CONCLUSIONS AND LIMITATIONS

### 4.1 CONCLUSIONS

The KP-FHR fuel qualification methodology is based on several decades of U.S. and international operating experience and testing with TRISO fuel particles and 20 years of extensive AGR fuel development and testing. The qualification methodology is informed by a PIRT review of the KP-FHR fuel with external subject matter experts and the implementation of recommendations from the PIRT. The KP-FHR fuel design specification, manufacturing, and quality control is developed to ensure that the KP-FHR fuel will be equivalent to the fuel tested in the AGR-1 and AGR-2 irradiation tests. The methodology also requires comparison of the actual fuel operating envelope with the fuel qualification envelope to ensure that the expected normal operation and LBEs are within the limits of the fuel qualification. The methodology includes a fuel laboratory testing program which demonstrates the acceptability of the KP-FHR fuel pebble, including mechanical, tribology, buoyancy, and material compatibility testing. Additionally, a fuel irradiation test may be utilized to expand the qualification limits defined by the AGR program. The acceptability of the KP-FHR fuel pebble form will be confirmed during startup and initial operation of the first KP-FHR and in the fuel surveillance program. During plant operation, an ongoing fuel surveillance program confirms that the fuel continues to perform as designed.

The results of fuel qualification laboratory testing described in Section 3.6 of this report will be submitted as part of the safety analysis reports for licensing applications under 10 CFR 50 or 10 CFR 52 documenting successful completion prior to startup of the initial KP-FHR. If fuel pebble irradiation is performed in a non KP-FHR test facility as described in Section 3.7.1, the results will also be included as part of the safety analysis reports for licensing applications under 10 CFR 50 or 10 CFR 52 documenting successful completion prior to startup of the initial KP-FHR.

A flow chart depicting the overall fuel qualification and licensing process is provided in Figure 4-1. The basis for the acceptability of the fuel qualification is summarized in Table 4-1.

### 4.2 LIMITATIONS

Kairos is requesting NRC approval of the fuel qualification methodology described in Sections 3.6 and 3.7 of this topical report for use by applicants for licenses of a KP-FHR under 10 CFR 50 and 10 CFR 52, and expects that, when the acceptance criteria for the qualification items in Table 4-1 are met, the fuel described in this report is qualified for use in the KP-FHR.

The use of this fuel qualification methodology for the KP-FHR TRISO fuel is subject to the following limitations:

1. The design of the annular pebble, TRISO particle based fuel and the KP-FHR design overview are as described in Section 1.1.2, including the presence of a Flibe primary coolant.
2. Operating and transient conditions for the KP-FHR are demonstrated in safety analysis reports submitted with license applications under 10 CFR 50 and 10 CFR 52 to remain within the fuel qualification envelope values specified in Table 3-11, which is based on the AGR program.
3. If the fuel qualification envelope is to be extended beyond the AGR-2 based limits, an irradiation test program will be conducted as described in Section 3.7.1.

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4. Demonstration that the conditions and limitations of the TRISO Topical Report Safety Evaluation Report (Reference 13) are met for the KP-FHR fuel design.
5. Future license applications for commercial electric power KP-FHRs will include justification (testing or analysis based on an approved methodology) of the applicability of this methodology during rapid reactor transient events for irradiated fuel.
6. Future license applications for commercial electric power KP-FHRs will include additional justification (testing or analysis based on an approved methodology) that Flibe does not adversely impact irradiated fuel pebble buoyancy.
7. This methodology applies only to KP-FHRs with a safety-related positive flux rate trip.

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**Table 1-1. KP-FHR Design Overview Description**

Parameter	Description / Value	
Reactor Type	Non-Power Test Fluoride-Salt-cooled High Temperature Reactor (FHR)	Commercial Electric Power Fluoride-Salt-cooled High Temperature Reactor (FHR)
Core Configuration	Pebble bed core, graphite moderator/reflector, and enriched Flibe molten salt coolant	Pebble bed core, graphite moderator/reflector, and enriched Flibe molten salt coolant
Refueling Intervals	Online and continuous with multiple passes through the core	Online and continuous with multiple passes through the core
Reactor Vessel Size	3 m diameter, 4.4 m height, 6 cm wall thickness	4 m diameter, 6 m height, 6 cm wall thickness
Reactor Thermal / Electric Power	35 MWth/Not applicable	320 MWth / 140 MWe
Reactor Operating Pressure	<0.2 MPa	<0.2 MPa
Core Inlet / Outlet Temperature	550°C / 650°C	550°C / 650°C
Reactor Coolant Flow Rate	210 kg/sec (~0.13 m/s)	1200-1400 kg/sec (~0.11-0.15 m/s)

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**Table 2-1. Comparison of KP-FHR TRISO Fuel vs. Traditional TRISO Fuel**

Parameter	Traditional HTGR Fuel Pebble and TRISO Particle	KP-FHR Annular Fuel Pebble and TRISO Particle
Pebble Diameter (mm)	60	40
Inner Core Diameter (mm)	N/A	[[
Fuel Region (mm)	50 (diameter)	
Fuel Free Outer Shell Thickness (mm)	5	
Particle Packing Fraction	7-12%	]]
Number of TRISO Particles	9,500 - 16,400	~16,000
Fuel Kernel Material	UO <sub>2</sub>	UCO
Kernel Diameter (μm)	500	425
Buffer Thickness (μm)	90	100
IPyC Thickness (μm)	40	40
SiC Thickness (μm)	35	35
OPyC Thickness (μm)	40	40

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**Table 2-2. Primary functions of TRISO fuel particle components**

Fuel System Component	Diameter (Thickness)	Purpose
UCO Kernel $\text{UO}_2 + \text{UC} + \text{UC}_2$	425 $\mu\text{m}$	The kernel contains the fissile material. The addition of a limited amount of UC + $\text{UC}_2$ suppresses CO production mitigating kernel migration, particle over-pressure, and corrosion of the SiC layer. Oxygen remains sufficient to oxidize fission products that would otherwise diffuse through the IPyC and attack SiC in the higher mobility carbide form.
Porous Carbon Buffer Layer	625 $\mu\text{m}$ (100 $\mu\text{m}$ )	The porous carbon buffer layer provides void volume to accommodate fission product gases limiting pressure as burnup increases. This layer mechanically de-couples the kernel from the outer coating layers and accommodates fuel kernel swelling. The layer also protects the IPyC from damage by fission product recoil.
IPyC Layer	705 $\mu\text{m}$ (40 $\mu\text{m}$ )	The IPyC layer protects the kernel from chlorine attack during SiC deposition in the manufacturing process. This layer introduces a compressive stress on the SiC layer that reduces SiC deformation and the risk of SiC layer failure during irradiation. The layer serves to protect the SiC from fission product attack. This coating layer is considered to be the secondary structural and fission product gas barrier after the SiC layer.
SiC Layer	775 $\mu\text{m}$ (35 $\mu\text{m}$ )	The SiC layer is the primary structural layer and fission product barrier. This layer is a diffusion barrier to mobile metallic and gaseous fission products.
OPyC Layer	855 $\mu\text{m}$ (40 $\mu\text{m}$ )	The OPyC layer protects the SiC layer during manufacture. This layer introduces a compressive stress on the SiC layer during irradiation that reduces SiC deformation and the risk of SiC layer failure. This coating layer is considered to be a secondary structural and fission product gas barrier after the SiC layer. This layer also separates the SiC layer from the carbon over-coat. A weak bond is formed between the carbon over-coat material and the OPyC layer facilitating the manufacture of the fuel region of the annular pebble.
Pebble - Particle Carbon Over-Coat	[[ ]]	The TRISO particle overcoat with carbon matrix material prevents particle-to-particle contact during manufacture. The overcoat also facilitates obtaining the nominal packing fraction in the pebble fuel region during manufacture.

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**Table 3-1. German, Chinese, Japanese, and U.S. Fuel Irradiation Tests**

Experiment /Reactor	TRISO Kernel Type	Reactor	Time (EFPD)	Peak Temperature (°C)	Peak Burnup (%FIMA)	Peak Fluence <sup>1</sup> (x10 <sup>25</sup> n/m <sup>2</sup> , E>0.1MeV)	Average Particle Power (mW)
German	UO <sub>2</sub> , (U, Th)O <sub>2</sub>	R-2, BR-2, HFR, DIDO, SILOE	232 - 651	800 - 1320	6.7 - 15.6	0.2 - 8.5	100 - 250
Chinese	UO <sub>2</sub>	HFR	355	1017 - 1067	9 - 11	3.8 - 4.9	150 - 250
Japanese	UO <sub>2</sub>	HFIR	89	1156	6.7	2.8	550
U.S. Legacy	UCO, UO <sub>2</sub> , UC <sub>2</sub> , (U,Th)O <sub>2</sub> , (U,Th)C <sub>2</sub>	GETR, R-2, ORR, HFIR	64 - 517	915 - 1350	12 - 80	2.1 - 11.5	100 - 400
U.S. AGR	UCO, UO <sub>2</sub>	ATR	361 - 620	800 - 1400	13.2 - 19.6	3.8 - 6.1	18 - 247
Aggregate Range of Legacy Testing	--	--	64 - 651	800 - 1400	6.7 - 19.6	0.2 - 11.5	55 - 550
Non-Power Test KP-FHR	UCO	KP-FHR	[[				
Commercial Electric Power KP-FHR	UCO	KP-FHR					]]

<sup>1</sup> Factor of 1.1 used to convert E >0.18 MeV to E >0.10 MeV



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**Table 3-2. TRISO Fuel Particle Phenomena Average Importance, Knowledge Level, and Rank**

ID	Sub-Component	Scenario	Phenomenon	Addressed by Listed Topical Report Section	Action
<b>Rank 1 – Phenomena with High Importance and Low Knowledge Level</b>					
None					
<b>Rank 2 – Phenomena with High Importance and Medium Knowledge Level</b>					
67	Fuel Kernel	NO/AOO	Gas phase diffusion	3.8.1 Fuel performance model, 3.2.2/3.2.5 AGR-1/AGR-2 data	[[
68	Fuel Kernel	NO/AOO	Condensed phase diffusion	3.8.1 Fuel performance model, 3.2.2/3.2.5 AGR-1/AGR-2 data	
47	Particle Coating Layers	PHSS	Cracking	3.6.1 Laboratory Program – Mechanical Tests	
73	Buffer Layer	NO/AOO	Bond strength to IPyC	3.8.1 Fuel performance model, 3.2.2 AGR-1/AGR-2 data	
27	IPyC Layer	MFG-Specs	Bonding strength to SiC	3.4 Fuel specification, manufacturing, and inspection	
28	IPyC Layer	MFG-Specs	Anisotropy	3.4 Fuel specification, manufacturing, and inspection	]]

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80	IPyC Layer	NO/AOO	Fast fluence	3.8.1 Fuel performance model, 3.2.2 AGR-1/AGR-2 data	[[
81	IPyC Layer	NO/AOO	Radiation induced, Thermal creep	3.8.1 Fuel performance model, 3.2.2 AGR-1/AGR-2 data	
82	IPyC Layer	NO/AOO	Anisotropy	3.8.1 Fuel performance model, 3.2.2 AGR-1/AGR-2 data	
83	IPyC Layer	NO/AOO	Dimensional change	3.8.1 Fuel performance model, 3.2.2 AGR-1/AGR-2 data	
84	IPyC Layer	NO/AOO	Debonding	3.8.1 Fuel performance model, 3.2.2 AGR-1/AGR-2 data	
85	IPyC Layer	NO/AOO	Stress state (compression/tension)	3.8.1 Fuel performance model, 3.2.2 AGR-1/AGR-2 data	
86	IPyC Layer	NO/AOO	Cracking	3.8.1 Fuel performance model, 3.2.2 AGR-1/AGR-2 data	
29	SiC Layer	MFG-Specs	Defects	3.4 Fuel specification, manufacturing, and inspection	]]
34	SiC Layer	MFG-Specs	Fracture strength	3.4 Fuel specification, manufacturing, and inspection	]]

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					[[
90	SiC Layer	NO/AOO	Cracking	3.8.1 Fuel performance model, 3.2.2 AGR-1/AGR-2 data	
93	SiC Layer	NO/AOO	Fission product corrosion	3.8.1 Fuel performance model, 3.2.2 AGR-1/AGR-2 data	
94	SiC Layer	NO/AOO	Gas phase diffusion	3.8.1 Fuel performance model, 3.2.2/3.2.5 AGR-1/AGR-2 data	
95	SiC Layer	NO/AOO	Condensed phase diffusion	3.8.1 Fuel performance model, 3.2.2/3.2.5 AGR-1/AGR-2 data	
160	SiC Layer	LBE	Cracking	3.8.1 Fuel performance model, 3.2.5 AGR-1/AGR-2 data	]]
<b>Rank 3 – Phenomena with Medium Importance and Low Knowledge Level</b>					
None					

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**Table 3-3. Pebble Phenomena Average Importance, Knowledge Level, and Rank**

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**Table 3-4. Limitations and Conditions from SER on TRISO Topical Report**

Limitation or Restriction	Kairos Power
<p>Limitation 1 – The scope of this [EPRI Topical Report] applies only to the UCO TRISO particles themselves. How the final fuel form is qualified and any impacts of the fuel form on the holistic fuel performance (for instance, any uranium contamination in the compact material) is the responsibility of the vendor or designer referencing this TR.</p>	<p>[[</p> <p>]]</p>

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Limitation or Restriction	Kairos Power
Limitation 2 – The [EPRI Topical Report – Reference 13] applies only to UCO TRISO particles that fall within the ranges discussed in Section 5.3 of the [EPRI Topical Report}. If an applicant chooses to use UO <sub>2</sub> /UC <sub>2</sub> ratios or burnup values that differ meaningfully from those used in the AGR program, the applicant must provide a justification for how the burnup and carbon content ratios conform to the performance ranges discussed in Section 5.3 of the TR.	[[
Condition 1 – An applicant or licensee referencing the [EPRI Topical Report – Reference 13] must evaluate any substantial discrepancies between their fuel particles and the TRISO particles used in the AGR program - specifically, reviewing the ranges specified in Table 5-6 for stress values to capture any effects from different kernel sizes to ensure the data in the TR remain applicable.	
Condition 2 – The performance limits in Table 6-6 and Figure 6-30 of the [EPRI Topical Report – Reference 13] are the result of different tests with distinct samples, not all of which had the maximum bounds occur during the same test. Applicants referencing the [EPRI Topical Report] must ensure that they either remain within the tested bounds or justify how their proposed operating conditions remain applicable.	]]



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**Table 3-5. Summary of AGR Irradiation Tests**

Test	Number of Particles (Packing Fraction)	Test Objective	Test Description
AGR-1	298,000 (36-37%)	Shakedown – Early fuel performance data to support fundamental understanding of relationship between fabrication, product properties, and irradiation performance. Particles for post-irradiation heating tests.	UCO particle compacts, baseline composite and three variations – two different IPyC and one different SiC coating conditions.
AGR-2	114,000 (37%)	Fuel performance demonstration – Particle variants investigation continued into fundamental understanding and heating tests, establish UCO performance margin.	UCO and UO <sub>2</sub> compacts made in larger size coatiers. UCO at high time-averaged temperature to demonstrate performance margin.
AGR-3/4	91,000 (37%)	Fission product transport – Provide data on fission gas release from failed particles, fission metal diffusion in kernels, and gas and metal diffusion in coatings for use in development of fission product transport models.	UCO particles irradiated with designed-to-fail particles to characterize fission product transport and release.
AGR-5/6	516,000 (25% and 38%)	Fuel qualification – Irradiation testing and post-irradiation heating of fuel particles in sufficient statistical quantities to demonstrate performance under normal operation and accident conditions.	Single UCO particle type used with best performance results from available AGR-1 and AGR-2 data.
AGR-7	54,000 (25%)	Fuel performance margin – demonstrate capability of fuel to exceed AGR-5/6 conditions in support of reactor design and licensing.	Same AGR-5/6 UCO particle type tested to conditions that exceed operating envelope expecting measurable level of failures to occur.

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**Table 3-6. Summary of AGR Irradiation Test Conditions (Reference 13, 54, 55)**

Test	Enrich. U-235 wt%	Time, EFPD	Peak Particle Power, mW	Ave. Particle Power, mW	Peak Burnup Compact, %FIMA	Ave. Burnup of All Compacts, %FIMA	Time-Ave. Peak Temp. Compact, °C	Time-Ave. Vol. Ave. Temp. of All Compacts, °C	Peak Fluence Compact <sup>1</sup> , x10 <sup>21</sup> n/cm <sup>2</sup> , E > 0.1 MeV	Ave. Fluence of All Compacts <sup>1</sup> , x10 <sup>21</sup> n/cm <sup>2</sup> , E > 0.1 MeV
AGR-1	19.7	620	104	56	19.6	16.7	1197	1044	4.7	3.8
AGR-2	14.0	559	155	73	13.2	11.1	1360	1142	3.8	3.2
AGR-3/4	19.7	369	98	65	15.3	11.9	1418	1087	5.8	4.4
AGR-5/6 <sup>1</sup>	15.5	361	247	107	15.3	10.6	1210	898	6.0	3.6
AGR-7 <sup>1</sup>	15.5	361	238	148	15.0	14.5	1405	1289	6.1	6.0

<sup>1</sup> Factor of 1.1 used to convert E > 0.18 MeV to E > 0.10 MeV

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**Table 3-7. Summary of AGR Safety Tests for Compacts with UCO TRISO Fuel Particles**

Test Temperature (°C)	Number of Compacts	Number of Particles	SiC Failures		TRISO Failures	
			Number of Failures	95% Confidence	Number of Failures	95% Confidence
AGR-1						
1600	8	33100	3	$\leq 2.4 \times 10^{-4}$	0	$\leq 9.1 \times 10^{-5}$
1700	3	12400	7	$\leq 1.1 \times 10^{-3}$	0	$\leq 2.5 \times 10^{-4}$
1800	4	16500	23	$\leq 2.0 \times 10^{-3}$	2	$\leq 3.9 \times 10^{-4}$
AGR-2						
1600	4	12704	0	$\leq 2.4 \times 10^{-4}$	0	$\leq 2.4 \times 10^{-4}$
1800	3	9528	1	$\leq 5.0 \times 10^{-4}$	1	$\leq 5.0 \times 10^{-4}$
AGR-1 and AGR-2						
1600	12	45804	3	$\leq 1.7 \times 10^{-4}$	0	$\leq 6.6 \times 10^{-5}$
1800	7	26028	24	$\leq 1.3 \times 10^{-3}$	3	$\leq 3.0 \times 10^{-4}$

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**Table 3-8. Reactivity Transient Tests of TRISO Fuel Particles and Fuel Forms (References 57, 58)**

Reactor (Country)	Fuel Type	Energy Deposition (J/g- UO <sub>2</sub> )	Pulse Width (ms)
NSRR (Japan)	Element (compacts) and loose particles	500 - 2300	~5
NSRR (Japan)	Loose particles	500 - 1700	~5
Hydra (Russia)	Element (compacts) and loose particles	100 - 1700	1-2
IGR (Russia)	Element (pebble)	>10,000	700-30,000

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**Table 3-9. Japanese NSRR Tests and Stages of Fuel Particle Failure (Reference 58)**

Energy Deposition (J/g-UO <sub>2</sub> )	Estimated Peak Fuel Temperature (°C)	Measured TRISO Fuel Particle Failure Fraction
578	1237	0
695	1557	0
1053	2387	0
1254	2727	0
1436	2967 <sup>1</sup>	0.02
1702	3377	0.74
1642	3217	0.37
1869	3677	0.97

<sup>1</sup> UO<sub>2</sub> melting point is ~2850°C (Reference 60 and 61)



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**Table 3-10. Summary of German Furnace Safety Testing of LEU UO<sub>2</sub> TRISO Fuel Particles in 60mm Diameter Spherical Fuel Elements**

Test Temperature (°C)	Number of Compacts	Number of Particles	Number of Fuel Particle Failures	Failure Fraction with 95% Confidence
Isothermal testing at the Julich Research Center of AVR fuel pebbles 71/22, 82/9, 82/20, 88/15, 88/33, and irradiation test pebbles HFR-K3/1, FRJ2-K13/2 and 4				
1600	8	131,200	0	---
Simulated accident testing at the Julich Research Center of AVR GLE fuel pebbles AVR 85/18, 89/13, 90/2 (two failures), 90/5 and 90/20 (three failures)				
1620	5	82,000	5	---
Isothermal testing at the Institute of Transuranium (ITU) of AVR fuel pebble 74/18, and irradiation test pebbles HFR-K6/2,3 and HFR-EU1bis/1, 3, 4				
1600	6	74,280	0	---
Total Tested				
1600	19	287,480	5	$\leq 3.7 \times 10^{-5}$

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**Table 3-11. Fuel Qualification Envelope and Anticipated Operating Envelope for KP-FHR for Normal Operation and Postulated Events**

Parameter	TRISO Particle Qualification Envelope	Anticipated Commercial Electric Power KP-FHR Conditions
Normal Operation		
Peak SiC Layer Temperature (°C)	1360	[[
Burnup (%FIMA)	13.2	
Peak Particle Power (mW)	155	
Peak Fluence ( $\times 10^{25} \text{n/m}^2$ , $E > 0.1 \text{ MeV}$ )	$3.8^2$	]]
Postulated Events		
Peak SiC Layer Temperature <sup>3</sup> (°C)	1600	[[
Peak Kernel Temperature <sup>4</sup> (°C)	2350	]]

Notes:

1. The fuel irradiation test program described in Section 3.7 will demonstrate the acceptability of operating at higher peak particle powers.
2. Factor of 1.1 used to convert  $E > 0.18 \text{ MeV}$  to  $E > 0.10 \text{ MeV}$ .
3. Burnup and fluence are long term steady state irradiation effects. AGR safety testing demonstrated temperature and time-at-temperature are the predominant factors in short term reactor accidents.
4. The melting point of the fuel kernel should not be exceeded in a reactor transient (Reference 73).

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**Table 3-12. HALEU TRISO Fuel Particle Lot Specifications**

Property	Specified Mean Value
Kernel diameter ( $\mu\text{m}$ )	[[
Buffer thickness ( $\mu\text{m}$ )	
PyC thickness ( $\mu\text{m}$ )	
SiC thickness ( $\mu\text{m}$ )	
Kernel density ( $\text{g}/\text{cm}^3$ )	
Buffer density ( $\text{g}/\text{cm}^3$ )	
PyC density ( $\text{g}/\text{cm}^3$ )	
SiC density ( $\text{g}/\text{cm}^3$ )	
C/U atomic ratio	
O/U atomic ratio	
PyC BAF	
SiC aspect ratio	]]

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**Table 3-13. Allowable Contamination and Defect Fractions for KP-FHR TRISO Fuel Particles**

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**Table 3-14. Primary Functions of KP-FHR Annular Pebble Fuel Components**

Fuel System Component	Outer Diameter (Thickness)	Purpose
Low-density Carbon Core	[[	The low-density carbon matrix core reduces the pebble density ensuring pebble has net positive buoyancy in the Flibe coolant.
Fuel Region		The fuel region is a shell of carbon matrix material surrounding the porous carbon inner core. It is embedded with TRISO fuel particles at the nominal packing fraction. The fuel region locates fuel near the coolant decreasing the thermal resistance. This allows particle powers to be high while keeping fuel temperatures within limits.
Fuel-Free Carbon Outer Shell	]]	The fuel-free carbon outer shell protects the fuel region from mechanical damage and separates the fuel particles from the coolant.

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**Table 3-15. KP-FHR Annular Fuel Pebble Specifications**

[[

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**Table 3-16. Examples of KP-FHR Fuel Pebble Characteristics to be Measured during Inspection**

Source Material or Fabricated Component	Measured Characteristic in Inspection Program
U <sub>3</sub> O <sub>8</sub>	U-235 enrichment, uranium content, impurities, boron equivalent
Kernel	Diameter, density, sphericity, stoichiometry, impurities
TRISO fuel particle	Layer thickness, density, PyC anisotropy, SiC aspect ratio, surface and free uranium content
Natural and petroleum coke graphite	Density, grain size, surface area, impurities, boron equivalent
Binder material	Viscosity, molecular weight, melting point, impurities
Pebble fuel	Density, diameter, thermo-physical properties, mechanical properties, thickness of fuel free outer shell, surface defects, fraction of defective SiC layers (burn leach), uranium loading, uranium contamination in carbon matrix, ash and lithium content, boron equivalent

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**Table 3-17. TRISO Fuel Performance Parameters in KP-FHR Cores and AGR Irradiation Tests**

Timeframe	Non-Power Test KP-FHR	Commercial Electric Power KP-FHR	AGR-1 2009 AGR-2 2013
Reactor Power Level (MWth)	35	320	N/A
Fuel Pebbles in the Core	~36,000	~150,000	N/A
Particles per Fuel Pebble/Compact	~16,000	~16,000	4,100 / 3,200
Packing Fraction	0.37	0.37	0.37
U-235 Enrichment (wt%)	19.55	19.55	19.7 / 14.0
Peak Particle Power (mW)	155	270-310	104 / 155
Average Particle Power (mW)	73	135	56 / 73
Burnup (%FIMA)	~6	~19	19.6 / 13.2
Peak Fluence <sup>1</sup> ( $\times 10^{25}$ n/m <sup>2</sup> , E > 0.10 MeV)	[[	[[	4.7 / 3.8
Pebble Surface Temperature <sup>2</sup> (°C) – KP-FHR Core and AGR Time Averaged			800 - 1050 / 850 - 1200
Peak SiC Layer Temperature <sup>3</sup> (°C) – KP-FHR Core and AGR Time Averaged			800 - 1200 / 850 - 1350
Peak Kernel Temperature <sup>4</sup> (°C) – KP- FHR Core and AGR Time Averaged			800 - 1300 / 850 - 1450
Time (EFPD)	]]	]]	620 / 559

Notes:

1. A 1.1 factor is used to convert E>0.18 MeV to E>0.10 MeV.

2, 3, 4. The range of temperatures provided for the KP-FHRs include the core average to values including power peaking factors in comparison to AGR-1 and AGR-2 irradiation test time averaged temperatures. Core average fuel temperatures are expected to be similar to the time-average temperature over multiple passes of a pebble through the core, while the upper range values including power peaking factors would exceed time averaged temperatures over a pebble's lifetime.



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**Table 3-18. Acceptance Criteria for Fuel Laboratory Testing**

Laboratory Program Test	Measured Parameter	Acceptance Criteria
Compression	Crush strength	[[
Impact	Pebble fracture	
Tribology	Wear rate	
Buoyancy	Density (mass and volume), coefficient of thermal expansion	
Buoyancy	Flibe infiltration	]]

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Material Compatibility	Corrosion rate of the pebble carbon matrix in Flibe	[[
Material Compatibility	Corrosion rate of pebble carbon matrix in air	]]

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**Table 3-19. Irradiation Test Capsule Design Requirements**

Test Capsule Requirements	Description
Test Article Configuration and Number	The test will contain a number of fuel pebbles that results in a statistically significant number of TRISO fuel particles given constraints for geometric space, uniformity of fuel pebble temperatures, and gradients for key parameters.
Thermocouples	Measure the temperature at several locations in the specimen holder. Data is used to control the gas mixture and establish a temperature history for the test.
Flux Monitor	Measure the fluence at different axial locations to confirm neutronic model calculations.
Sweep Gas	Gas(es) are used to sweep through the capsule to vary the capsule gas mixture and permit radiation monitoring of radioactive gas release.
Gas Mixture Control	A gas mixture such as He/Ne is to be used to alter the effective thermal conductivity of the gas in the gas gap to enable temperature control in the test.
Single Compartment	The test articles are to be contained in a single compartment with sweep gas being common to all specimens. This maximizes the number of test articles and reduces test capsule complexity.
Gross Activity Detection	Gross activity of the sweep gas is to be measured to establish a test baseline and observe increases in activity with irradiation and due to particle failures.
Gamma Spectroscopy	Measurements of gaseous fission product isotope activity such as Kr and Xe are performed to establish release rate to birth rate ratios.
Data Acquisition System	Data from the listed instruments are to be collected with a data acquisition system in an electronic format that is transmittable.

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[[ **Table 3-20. Minimum sample size requirement for a 95% UCB**

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**Table 4-1. Acceptance Criteria for KP-FHR Initial Fuel Qualification**

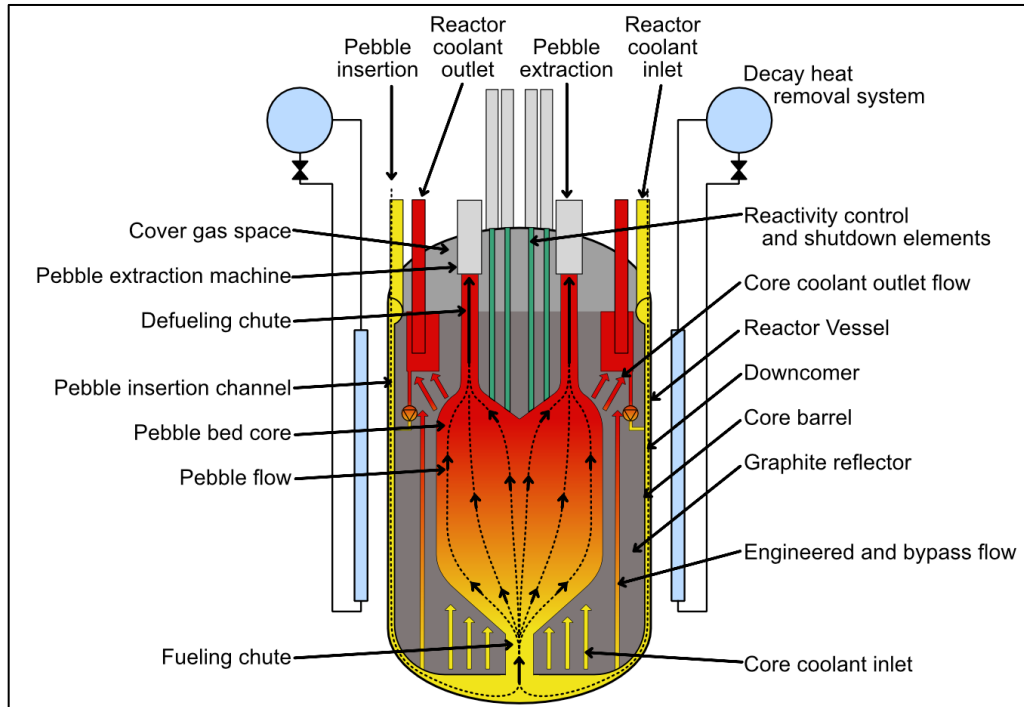
Item	Discussion	Acceptance Criteria
DOE AGR and Legacy Data	TRISO Experience from Germany, Japan, South Korea, China, U.S., and U.S. DOE AGR-1 and AGR-2.	[[
Fuel Pebble PIRT	Design-specific PIRT to identify phenomena related to fission product transport and release identified and ranked by importance and knowledge level. The PIRT is conducted with external expert input.	
Fuel Specification, Manufacturing, and Quality Control through Inspection	Fuel Specification derived from the AGR program.	]]

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Item	Discussion	Acceptance Criteria
Development of KP-FHR Fuel Qualification Envelope	Fuel qualification envelope defines the range of conditions associated with KP-FHR normal operation and accidents.	[[
Fuel Pebble Laboratory Testing	Mechanical, tribology, buoyancy, and material compatibility testing	]]

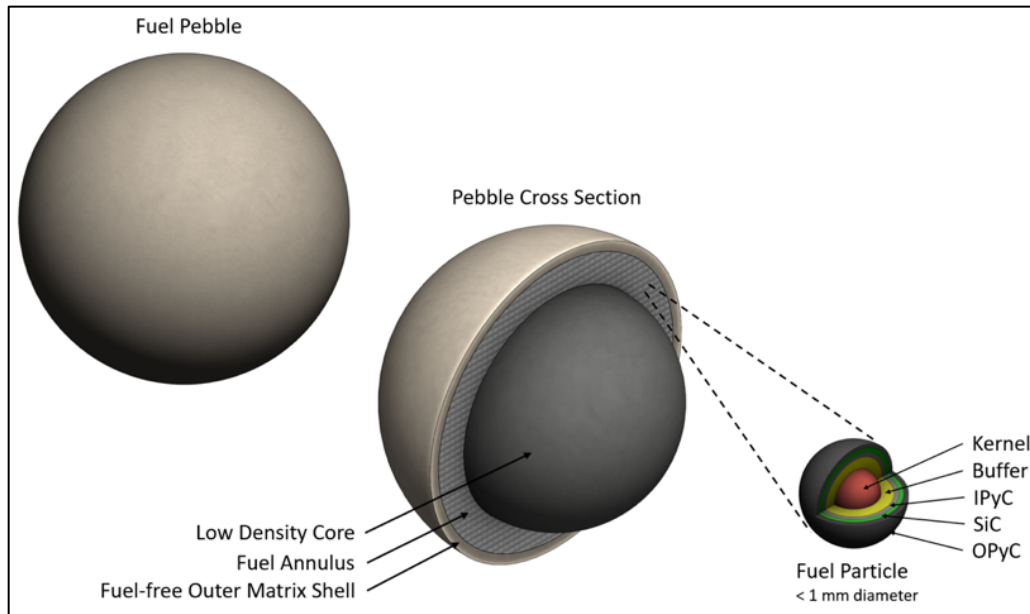


**Figure 1-1. Notional Configuration of a KP-FHR**

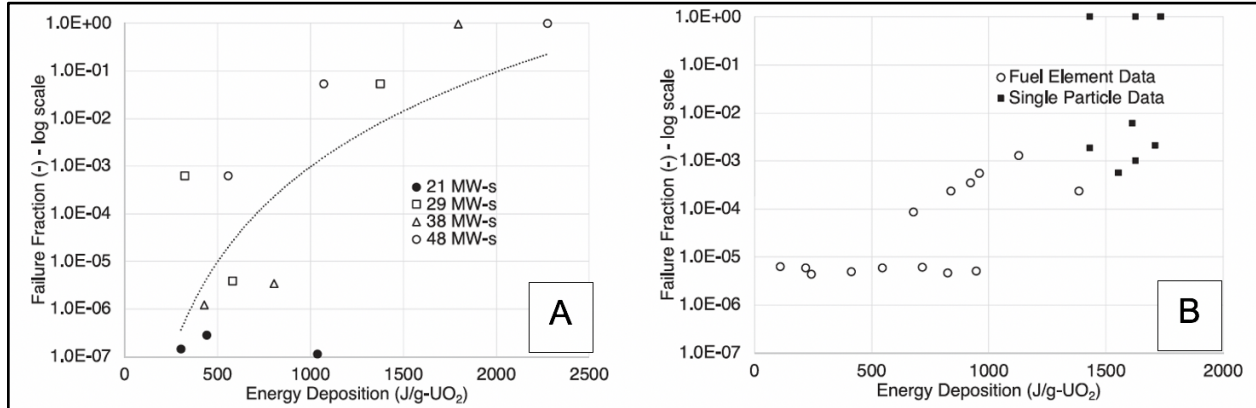




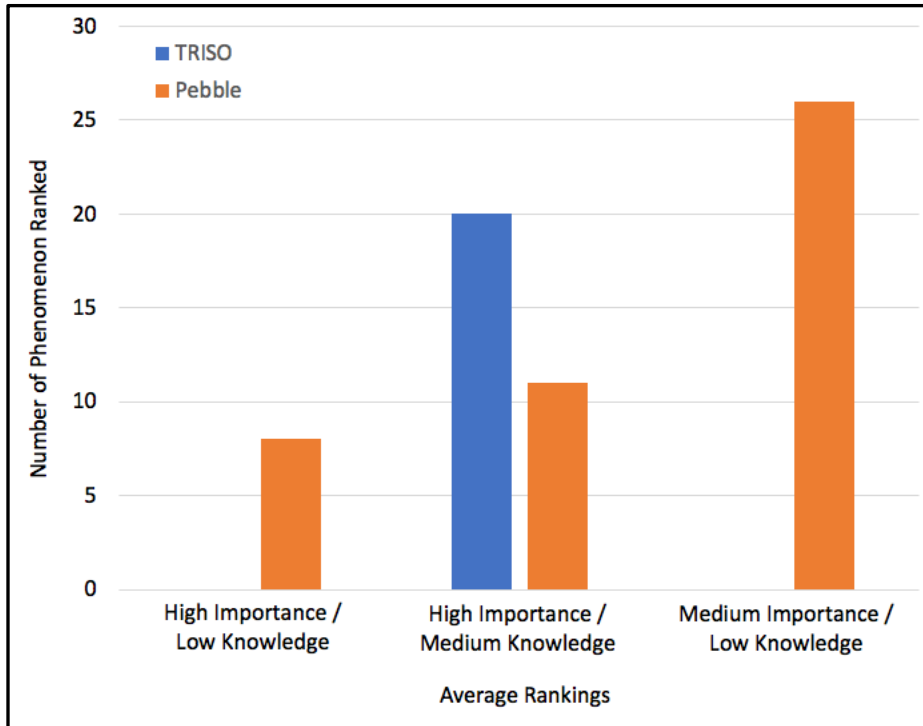
**Figure 2-1. Notional Design of TRISO Particles**



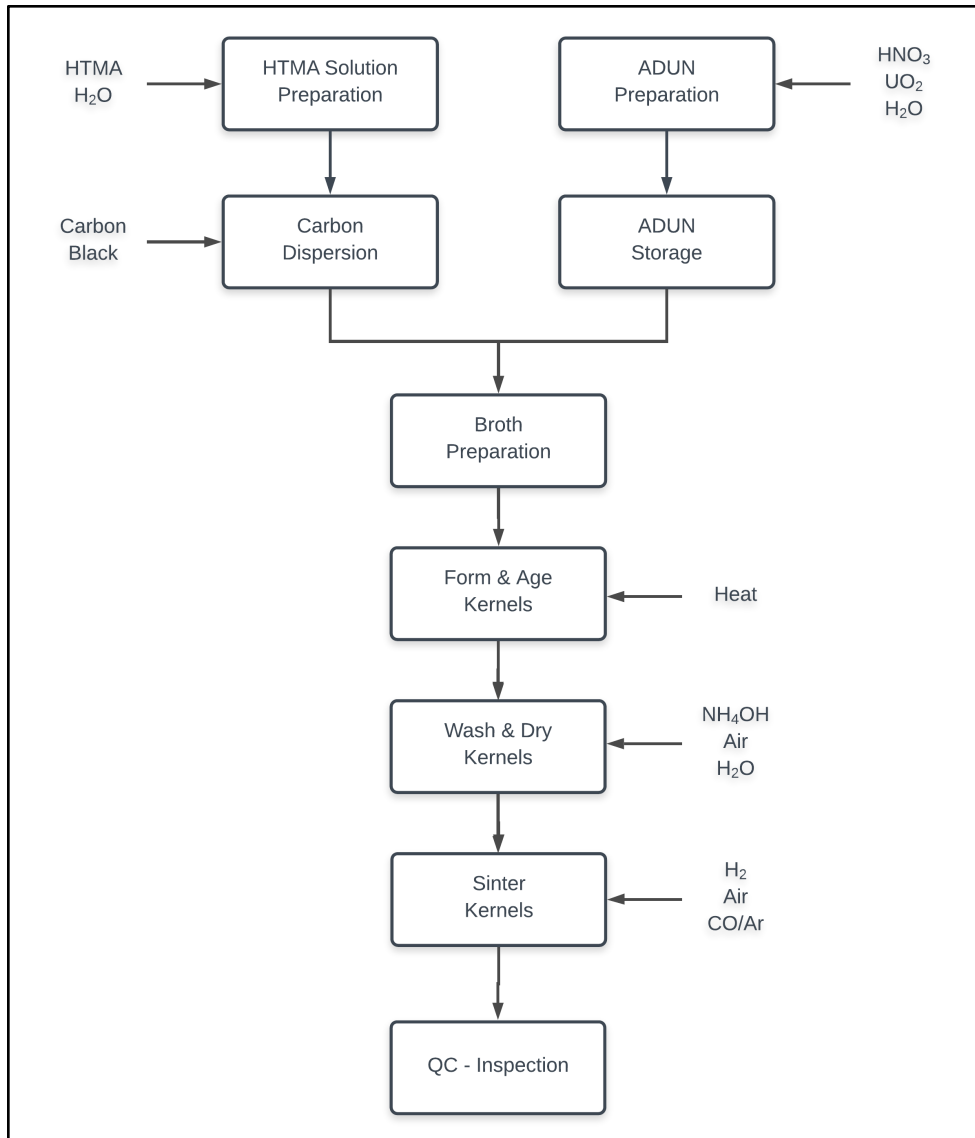
**Figure 3-1. TRISO Fuel Particle Failure Fractions and Energy Deposition in Reactivity Tests (Reference 59) A) Japanese NSRR tests, and B) Russian Hydra tests**



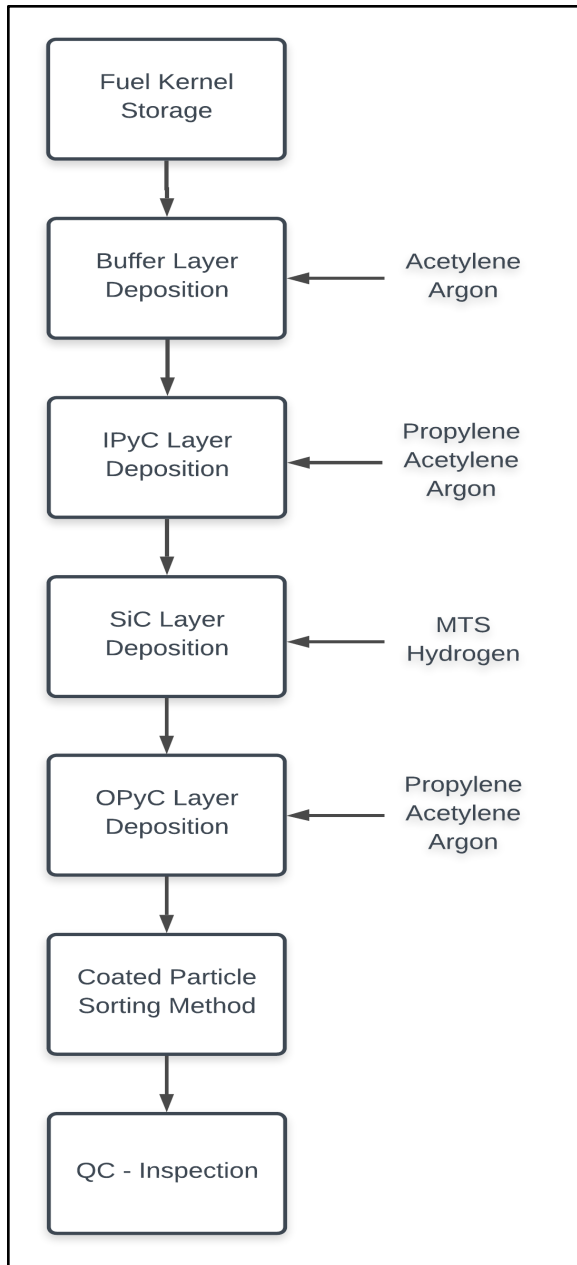
**Figure 3-2. PIRT Summary Results for Particles and Pebbles**



**Figure 3-3. Fuel Kernel Fabrication Process Flowchart**



**Figure 3-4. TRISO Fuel Particle Coating Process Flowchart**



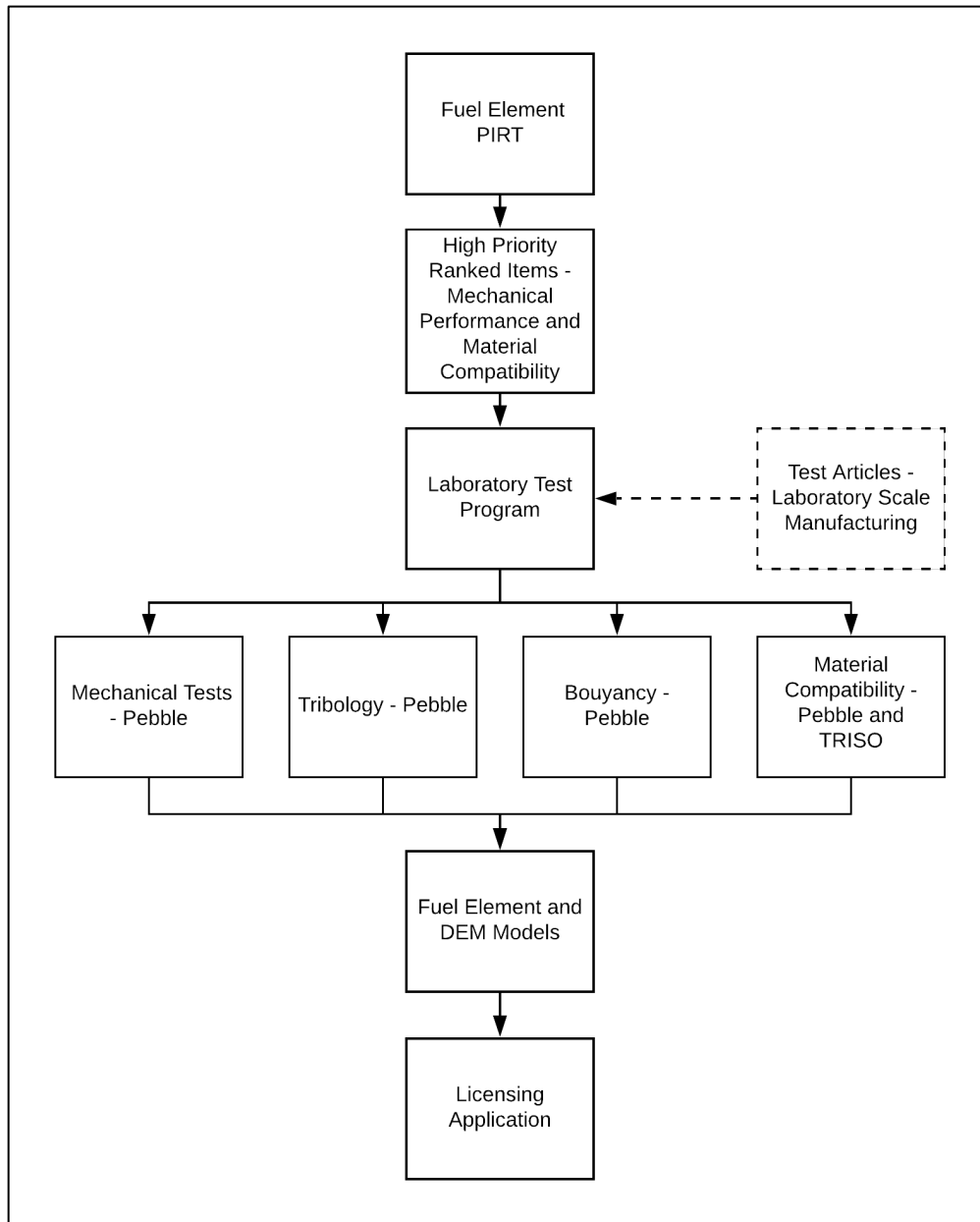
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**Figure 3-5. Annular Pebble fuel Fabrication Process Flowchart**

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**Figure 3-6. Laboratory Test Program**



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**Figure 3-7. Comparison of Non-Power Test and Commercial Electric Power KP-FHR Equilibrium Core TRISO Particle Conditions with the Fuel Qualification Envelope**



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**Figure 4-1. Integrated Fuel Qualification/Licensing Flowchart**

**Figure 4-2. Topical Report Methodology Relationships**

