

US DESIGN CERTIFICATION

PBMR FUEL PERFORMANCE ENVELOPE AND TEST PROGRAM

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ABSTRACT

This paper identifies the regulatory topics related to the fuel performance envelope and fuel test program for which U.S. Nuclear Regulatory Commission (NRC) feedback is desired during the pre-application review of the Pebble Bed Modular Reactor (PBMR). The regulatory foundation for review of PBMR fuel is summarized, compliance with the regulatory criteria is described, and specific issues for which feedback is requested are described. The anticipated PBMR operating envelope for fuel is defined, existing German irradiation and post-irradiation testing data are reviewed, and the PBMR test program intended to provide additional data in support of PBMR operation is described.

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ABBREVIATIONS

This list contains the abbreviations used in this document.

Abbreviation or Acronym	Definition
ASME	American Society of Mechanical Engineers
AVR	<i>Arbeitsgemeinschaft Versuchsreaktor</i> (Jointly-operated Prototype Reactor) in Jülich, Germany
BISO	Binary Coated Particle (Coated particles with two layers – low density pyrocarbon/dense pyrocarbon)
CBA	Core Barrel Assembly
CFR	Code of Federal Regulations
CRDM	Control Rod Drive Mechanism
CS	Core Structures
CSC	Core Structures Ceramics
CUD	Core Unloading Device
DBA	Design Basis Accident
DCA	Design Certification Application
DIDO	Heavy water (D ₂ O, or DDO) moderated test reactor in Jülich Germany
DLOFC	Depressurized Loss of Forced Cooling
ECCS	Emergency Core Cooling System
EPRI	Electric Power Research Institute
FIMA	Fissions per Initial Heavy Metal Atom
FRJ2	Forschungsreaktor Jülich No. 2: DIDO reactor in Jülich, Germany
FS	Fuel Sphere
GDC	General Design Criteria
GLE	Gepresst (pressed) Low Enriched – fuel spheres
HFR	High Flux Reactor in Petten, Netherlands
HTGR	High Temperature Gas-cooled Reactor
HTR	High Temperature Reactor
ILTI	Inner Low Temperature Isotropic pyrocarbon layer
KÜFA	<i>Kühlfingerapparat</i> (German for 'cold finger apparatus) – a facility for performing post irradiation heating tests of coated particle fuel in Jülich, Germany
LEU	Low Enriched Uranium
LWR	Light Water Reactor
MHTGR	Modular High Temperature Gas-cooled Reactor
MTR	Materials Test Reactor
n/a	not applicable
N/D	Not Determined
NRC	Nuclear Regulatory Commission
NUREG	NUclear REGulatory Commission Report
OLTI	Outer Low Temperature Isotropic pyrocarbon layer
PBBMR	Pebble Bed Modular Reactor

Abbreviation or Acronym	Definition
PCU	Power Conversion Unit
PIE	Post-irradiation Examination
PLOFC	Pressurized Loss of Forced Cooling
PyC	Dense pyrocarbon layers adjacent to the silicon carbide layer in a coated particle
RAI	Request for Additional Information
R/B	Release-to-birth
RCS	Reactivity Control System
ROT	Reactor Outlet Temperature
RPV	Reactor Pressure Vessel
RSS	Reserve Shutdown System
RU	Reactor Unit
SAFDL	Specified Acceptable Fuel Design Limits
SAS	Small Absorber Sphere
SL	Siloe Reactor in Grenoble
SRP	Standard Review Plan
THTR	Thorium High Temperature Reactor
TINTE	Time Dependent Neutronics and Temperatures
TRISO	Triple Coated Isotropic Particle (Coated particle with three types of layers – low density pyrocarbon/dense pyrocarbon/silicon carbide/dense pyrocarbon)

1. INTRODUCTION

The qualification of Pebble Bed Modular Reactor (PBMR) fuel is based on 'equivalence' to fuel manufactured and tested in Germany. 'Equivalence' requires use of the German specifications, the same critical manufacturing equipment and process steps, the same or equivalent direct materials, and the same or improved quality control. This paper provides an overview of the fuel design; the German manufacturing, operation, and testing experience; the application of that experience to the PBMR program; and the PBMR confirmatory testing program.

In coated particles, defects from manufacturing or failures from service are of two types:

- Exposed kernels – Particles with connected failures in all coating layers that release both gaseous and metallic fission products during normal operation or accident conditions.
- Silicon carbide defects – Particles with cracks or permeable regions of the silicon carbide layer, but that have at least one intact dense pyrocarbon layer. These particles retain gaseous fission products, but may release metallic fission products during normal operation or under accident conditions.

There are approximately 7×10^9 coated fuel particles in the reactor core. Typical German as-manufactured defective particle fractions, as measured by the burn-leach procedure¹, are on the order of 10^{-5} . Consequently, a few tens of thousands of as-manufactured defective particles are expected to be present in the core, and more particles may fail during normal operation and during transients. Additionally, small amounts of enriched uranium contamination on the outer pyrocarbon layer, as well as trace natural uranium contamination in the matrix exist as a result of the manufacturing process and trace uranium in the matrix material constituents. This contamination constitutes a base level of the free uranium fraction that will contribute to fission product release during normal operation and accidents. The burn-leach procedure does not distinguish between exposed kernels and silicon carbide defects. However, as discussed in paragraph 3.3.3, the German data on initial gaseous and metallic fission product release during post-irradiation heating tests provide a means of distinguishing and quantifying the two types of failed particles. Additional testing will be performed on PBMR fuel to confirm the applicability of the German data to the fuel to be manufactured by PBMR, and to strengthen the fuel performance database. The confidence values given in this report are the level of confidence that the failure or defect fraction indicated is not exceeded in the parent population from which the sample was drawn.

This paper is structured as follows:

- Chapter 2 summarizes the regulatory foundation for review of the fuel design;
- Chapter 3 describes the PBMR approach to compliance with the regulatory criteria;
- Chapter 4 lists the regulatory issues expected to be addressed during the pre-application review; and
- Chapter 5 summarizes the expectations of PBMR for the outcomes of the pre-application review of this paper.

¹ The burn-leach procedure is a fuel characterization method whereby loose particles or spheres are subjected to exposure to air at elevated temperatures (~ 800 °C) for an extended period (tens of hours), followed by repeated immersion in a boiling nitric acid solution for several hours at a time. This procedure removes the outer pyrocarbon on all particles and sufficient inner pyrocarbon on particles with exposed kernels or silicon carbide defects to allow dissolution and extraction of the uranium in the kernel. The resulting 'free uranium fraction' includes both particles with exposed kernels, and particles with silicon carbide defects as well as uranium contamination outside the silicon carbide layer of the intact particles.

1.1 SCOPE AND PURPOSE

The integrity of PBMR fuel is especially critical to the safe and efficient operation of the plant. While plant safety depends on many factors such as low power density, use of helium as coolant, and passive heat removal, the integrity of the coated fuel particles within the fuel spheres is critical, because the coated particles themselves provide the primary barrier to fission product release. Maintaining the integrity of the multi-layered fuel particles is primarily dependent on assurance that the quality and properties of the as-manufactured fuel complies with the fuel specification, and that the fuel service conditions (e.g., temperature, temperature gradient, burnup, and fast fluence) remain within an allowed performance envelope. The nature of PBMR coated particle fuel is substantially different than conventional metal-clad pin-type fuels. It should be noted that the normal operation temperature envelope is for very long-term, continuous operation. Relatively short-term (tens of hours) temperature transients can substantially exceed the steady state envelope without degrading the most limiting fuel particles. This is one of the key differentiating features that allow the PBMR design to obtain substantially improved safety without active safety systems.

The purpose of this paper is to summarize the PBMR approach to, and obtain the Nuclear Regulatory Commission's (NRC's) pre-application review comments on, the application of the German fuel performance database to PBMR and the PBMR confirmatory test program scope, such that the combination of the two data sets would support one of the key bases for PBMR licensing. Complete information required for the NRC's full review of PBMR fuel that is not directly related to the purpose of this paper will be provided in the Design Certification Application (DCA).

1.2 STATEMENT OF THE ISSUES

The PBMR approach to fuel design and qualification is based on the extensive and successful performance of the spherical fuel program in Germany. PBMR will produce fuel equivalent to the fuel produced in Germany to ensure that the German experience database remains applicable to PBMR. PBMR operating parameters have been selected to be within the envelope defined by German experience, to ensure consistent, predictable performance during PBMR operation. PBMR fuel manufacturing equivalence is established by using the same specifications, the same critical manufacturing process steps and equipment, the same or equivalent direct materials, and the same or improved quality controls and monitoring. Notwithstanding this approach, PBMR will perform additional confirmatory tests on fuel manufactured at a qualified PBMR manufacturing facility to (1) statistically strengthen the performance database and (2) demonstrate that PBMR-manufactured fuel does, in fact, perform equivalent to or better than the German fuel upon which the PBMR fuel design is based.

The primary issues on which PBMR wishes to receive NRC staff feedback prior to submission of the DCA are:

- the adequacy of the subsets of German data for manufacturing experience, normal operation irradiation, and transient/accident heatup conditions that are judged to be applicable to PBMR;
- the adequacy of PBMR tests to statistically strengthen the database for normal operation and transient conditions; and
- the activities necessary to assure that the German and PBMR data are qualified for use in the DCA.

1.3 SUMMARY OF PRE-APPLICATION OUTCOME OBJECTIVES

PBMR is especially interested in those issues which are likely to require extensive effort in preparing an acceptable DCA.

Therefore, for those issues, it is requested that NRC reviewers provide either:

- confirmation that PBMR plans for addressing the issues in paragraph 1.2 are generally acceptable; or
- identification of any additional information needs of the NRC or any areas in which the NRC believes that PBMR's plans, when fully executed, will not be sufficient to address applicable regulatory requirements and guidance.

The NRC staff is requested to identify:

- any special documentation or material related to the PBMR fuel performance envelope and test program that would have to be added to the DCA in order to prepare a complete DCA; and
- any policy issues for which Commission direction is required.

It is expected that workshops will be held after NRC staff review of this paper and that a revised paper will be submitted, upon which final NRC comments will be provided.

1.4 RELATIONSHIP TO OTHER PRE-APPLICATION FOCUS TOPICS/PAPERS

This paper refers to the VSOP99 and TINTE computer codes that are used to provide fuel service conditions (burnup, fast fluence, temperature) for normal operation and accident conditions respectively. While the basic functions of VSOP99 and TINTE are summarized herein, a more detailed description of the VSOP99 and TINTE codes and their verification and validation will be provided in a separate paper on verification and validation of systems response modelling codes.

The limiting accident fuel service conditions addressed in this paper are derived from the analysis of the Depressurized Loss of Forced Cooling (DLOFC) event. The methods to be used for identification of licensing basis events are described in a paper on licensing basis event selection. The DLOFC event was identified as DBA-6 in that paper.

Likewise, this paper describes the use of existing German fuel manufacturing, irradiation, and post-irradiation testing data to develop expected fractions for the two particle failure modes described earlier. These fractions will be used as input to fuel response analysis methods (NOBLEG, GETTER) for determining release from the fuel during normal operation, anticipated operational occurrences, and Design Basis Accidents (DBAs). The verification and validation of these codes will be described in a paper on fuel response modelling. An overview of the mechanistic source term approach derived from the German fuel performance data and the fuel response codes will be provided in a separate paper on radionuclide release from the fuel.

2. REGULATORY FOUNDATION

2.1 NRC REGULATIONS

Regulations related to the scope of this document are codified primarily in the General Design Criteria (GDC) contained in Appendix A to 10 CFR Part 50, in 10 CFR §50.46, acceptance criteria for emergency core cooling systems for light-water nuclear reactors, in the quality assurance requirements of 10 CFR Part 50 Appendix B, and in 10 CFR §52.47 contents of applications for design certification.

The GDC establish minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design to existing conventional plants. The GDC were developed specifically for water-cooled designs, and are not requirements for other types of reactors. Nevertheless, as discussed in the introduction to Appendix A to 10 CFR Part 50, the GDC are considered to be generally applicable to other types of nuclear power units, and are intended to provide guidance in establishing the principal design criteria for such other units.

The GDC that provide some guidance relevant to the scope of this document are GDC 10 and 35. There are other GDC that pertain to the reactor core, such as GDC 11, 12 and 27, but these GDC do not directly pertain to the performance of fuel spheres (which is the subject of this paper).

In summary, GDC 10 and 35 contain the following requirements:

- GDC 10, *Reactor design*, states that 'the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.'
- GDC 35, *Emergency core cooling*, states that 'an emergency core cooling system shall be provided 'such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.'

Similarly, 10 CFR §50.46 requires light-water power reactor fuelled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding to be provided with an Emergency Core Cooling System (ECCS) that is designed so that its calculated cooling performance following postulated loss-of-coolant accidents conforms to specified criteria regarding peak cladding temperature, maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long-term cooling.

Due to design differences between water-cooled and gas-cooled reactors, the ECCS design requirements in GDC 35 and 10 CFR §50.46, other than those for coolable core geometry, are not applicable to the PBMR.

10 CFR Part 50 Appendix B is applicable to the production of fuel performance data via the irradiation and Post-irradiation Examination (PIE) and testing of fuel. As discussed in Chapter 3, PBMR intends to utilize existing data from German experiments as well as data developed for the PBMR demonstration power plant. While these data were not or will not necessarily be developed under an explicit Appendix B compliant quality assurance program, PBMR intends to demonstrate that the production of the data satisfies the intent of 10 CFR Part 50 Appendix B.

10 CFR §52.47 (b)(2) provides guidance on the content of the design certification application for designs which differ significantly from the Light Water Reactor (LWR) designs described in paragraph (b)(1), or utilize simplified, inherent, passive, or other innovative means to accomplish their safety functions. It states the need for a combination of analysis and test programs to demonstrate the performance of safety features and assure that sufficient data exist to assess the analytical tools used for safety analyses.

2.2 NRC POLICY STATEMENTS

NRC policy statements on nuclear fuel only address mixed oxide fuel and fuel using high enriched uranium. They do not address the type of fuel proposed for the PBMR, and do not address testing or monitoring of fuel. The NRC policy statement on regulation of advanced nuclear power plants does not explicitly address nuclear fuel. However, several of the attributes that should be considered, according to the policy statement, are enabled in the PBMR design by the characteristics of the coated particle fuel.

These attributes are:

- Highly reliable and less complex shutdown and decay heat removal systems.
- Longer time constants.
- Simplified safety systems.
- Minimized potential for severe accidents.
- Reduced potential radiation exposures to plant personnel.

2.3 NRC GUIDANCE

The NRC has established detailed guidance on fuel performance in NUREG-0800, Standard Review Plan (SRP) 4.2, Fuel System Design [1]. Also, NUREG-1338, Pre-application Safety Evaluation Report for the Modular High-Temperature Gas-Cooled Reactor [2] was issued in draft form by the NRC for the Modular High Temperature Gas-cooled Reactor (MHTGR) conceptual design. The initial draft of NUREG-1338 (published in March 1989) is instructive on how the NRC has applied SRP 4.2 to High Temperature Gas-cooled Reactors (HTGRs)². While this report was with regard to a UCO fissile/ThO₂ fertile particle system in a prismatic core, the NRC conclusions are for the most part applicable to the PBMR fuel.

The purpose of the fuel system safety review under SRP 4.2 is to provide assurance that the fuel design meets the requirements of GDC 10 and 35, and the core coolability requirements of 10 CFR §50.46. To this end, SRP 4.2 contains guidance on Specified Acceptable Fuel Design Limits (SAFDLs) that ensure that LWR fuel is not damaged as a result of normal operation and anticipated operational occurrences, that fuel damage is never so severe as to prevent control rod insertion when it is required, that the number of fuel rod failures is not underestimated for postulated accidents, and that coolability is always maintained. The SAFDL objectives were developed for water-cooled reactors, and no clear acceptance criteria exist for HTGR fuel in the current regulatory framework. In the PBMR design, no control rods are located in the fuelled region of the reactor, so control rod insertion is not affected by the condition of the fuel.

² References to specific page numbers in NUREG-1338 that are called out in paragraph 2.3 are to the March 1989 draft version of the NUREG.

To demonstrate that the SAFDLs have been established and satisfied, SRP 4.2 states that the NRC staff will review (1) the design bases for the fuel; (2) description and design drawings for the fuel; (3) the evaluation of the design of the fuel; and (4) plans for testing, inspection, and surveillance of the fuel. Each of these four areas is discussed in the following paragraphs.

2.3.1 Fuel Design Bases Acceptance Criteria

The SRP indicates that fuel design bases should reflect the SAFDL objectives described in paragraph 2.3. To satisfy these objectives, acceptance criteria are needed for fuel damage, fuel failure, and fuel coolability.

- a. **Fuel Damage** – To meet the requirements of GDC 10 as it relates to SAFDL for normal operation, including anticipated operational occurrences, SRP 4.2 states that fuel damage criteria should be given for all known damage mechanisms, including:
- Stress, strain, or loading limits for spacer grids, guide tubes, thimbles, fuel rods, control rods, channel boxes, and other fuel system structural members should be provided. If stress limits are obtained by methods other than those specified, Section III of the ASME Code the proposed limits should be justified.
 - The cumulative number of strain fatigue cycles on the structural members should be significantly less than the design fatigue lifetime, which should be based on appropriate data and include a safety factor of 2 on stress amplitude or a safety factor of 20 on the number of cycles.
 - Fretting wear at contact points to structural members should be limited, and allowable fretting wear should be stated in the safety analysis.
 - Oxidation, hydriding, and the buildup of corrosion products (crud) should be limited, and allowable oxidation, hydriding, and crud levels should be discussed in the safety analysis and shown to be acceptable. These types of mechanisms are not present in the PBMR. However, NUREG-1338 (pages 4-5 to 4-6) indicates that analogous mechanisms, such as chemical decomposition failure modes applicable to HTGR fuel and the potential effects of water intrusion into the system, should be discussed and evaluated.
 - Dimensional changes such as rod bowing or irradiation growth of fuel rods, control rods, and guide tubes should be included in the analysis to establish operational tolerances.
 - Fuel and burnable poison rod internal gas pressures should remain below the nominal system pressure during normal operation unless otherwise justified.
 - Worst-case hydraulic loads for normal operation should not exceed the holddown capability of the fuel assembly (either gravity or holddown springs).
 - Control rod reactivity should be maintained.

Except for control rods, these types of components are not contained in the PBMR, and the control rods are not located in the fuelled region of the reactor. However, NUREG-1338 (page 4-4) indicates that the essential elements of the Triple Coated Isotropic Particle (TRISO) coating should be evaluated, and the structural function of each layer of the coating should be analyzed. In NUREG-1338 (page 4-6), the NRC indicated that the acceptance criteria for the HTGR fuel design should be 'at least the functional equivalent of the LWR fuel-acceptance criteria in accordance with Section 4.2 of the Standard Review Plan.' Relevant fuel damage mechanisms will be addressed in

the DCA, but the list is expected to be considerably shorter and simpler due to the relative simplicity of the PBMR fuel spheres in comparison to an LWR fuel assembly.

- b. **Fuel Failure** – To meet the requirements of GDC 10 as it relates to SAFDL for normal operation, including anticipated operational occurrences (as well as 10 CFR Part 100 as it relates to fission product releases for postulated accidents), page 4.2-8 of the SRP states that fuel rod failure criteria should be given for all known fuel failure mechanisms. The SRP lists eight failure modes for water cooled reactor fuel: (1) internal hydriding; (2) cladding collapse; (3) fretting of cladding; (4) overheating of cladding; (5) overheating of fuel pellets; (6) excessive fuel enthalpy; (7) bursting; and, (8) mechanical fracturing.

Because no metal cladding is used in the design of the PBMR fuel, internal hydriding, cladding collapse, fretting of cladding, and overheating of cladding failure modes are not applicable to the PBMR. However, in NUREG-1338 (page 4-4), the NRC examined failure modes for TRISO-coated fuel particles. Failure modes examined were different than those discussed above for conventional water-cooled reactor fuel, and included the following:

- pressure-induced failures;
- irradiation-induced failures;
- failures due to thermal decomposition of silicon carbide at elevated temperatures; and
- failures due to internal corrosion mechanisms.

Due to the very large number of fuel particles employed in HTGR designs, NUREG-1338 (page 4-6) states that use of statistical methods would be emphasized in evaluating such fuel designs.

- c. **Fuel Coolability** – SRP 4.2 (page 4.2-10) states that fuel assemblies should retain coolability including retaining rod-bundle geometry with adequate coolant channels to permit removal of residual heat. Reduction of coolability can result from cladding embrittlement, violent expulsion of fuel, generalized cladding melting, gross structural deformation, and extreme coplanar fuel rod ballooning. The PBMR fuel design is sufficiently different from traditional designs that the majority of fuel coolability considerations typically used for water-cooled reactors would not apply to PBMR.

2.3.2 Fuel Description and Design Drawings

The SRP (page 4.2-12) states that the NRC should be provided with a description and design drawings that are complete enough to provide an accurate representation of the fuel. This includes comprehensive dimensional and metallurgical information regarding the cladding; fuel pellet data including dimensions, roughness, density, re-sintering data, burnable poison content; internal void volume and fill gas type and pressure; enrichment data; hydraulic diameter; coolant design pressure; and burnup limit. Equivalent information will be provided in the DCA.

2.3.3 Fuel Design Evaluation

The SRP (page 4.2-14) states that the NRC will review the methods of demonstrating that the design bases are met. To ensure that the design bases are met, the NRC will examine (a) operating experience with the fuel and other similar designs, (b) prototype testing, and (c) analytical predictions. NUREG-1338 (page 4-7) also stresses that adequacy of the

technology development plan for fuel development is an essential element for staff acceptance of an HTGR.

- a. **Operating Experience** – Page 4.2-14 of the SRP states that the NRC will review operating experience with fuel systems of the same or similar design. The SRP states that such experience should be described in detail, including the maximum burnup achieved. The SRP further states that the NRC values actual operating experience over prototype testing or analytical predictions.
- b. **Prototype Testing** – Page 4.2-14 of SRP 4.2 states that the NRC will look to prototype testing to demonstrate adherence to the fuel design bases. Prototype testing typically includes both out-of-reactor and in-reactor testing.
 - The SRP states that out-of-reactor tests should be performed when practical to determine the characteristics of the new design. The SRP does not contain definitive requirements regarding design features that should be tested prior to irradiation. However, it states that out-of-reactor tests have been performed for past designs of fuels assembly structural components and hydraulic characteristics.
 - The SRP states that in-reactor testing of design features and lead-assembly irradiation of whole assemblies of a new fuel design will be reviewed by the NRC. Of particular interest is the maximum burnup experience achieved in in-reactor prototype testing in relation to the specified maximum burnup limit for the new design.

For the HTGR, NUREG-1338 (page 4-9) states that statistically low failure rates assumed in the fuel safety analysis will require a rigorous research and development program that complies with a systematic statistical approach commensurate with the number of parameters and the required accuracy.

- c. **Analytical Predictions** – SRP 4.2 acknowledges that some design bases and related parameters can only be evaluated analytically. Page 4.2-15 of SRP 4.2 provides a list of parameters that NRC will review, including analytical models for fuel temperatures (stored energy), densification effects, fuel rod bowing, structural deformation, rupture and flow blockage, fuel rod pressure, metal/water reaction rate, and fission product inventory. In NUREG-1338 (pages 4-6 and 4-8), the NRC relied upon NUREG-0111, *Evaluation of High-Temperature Gas-Cooled Reactor Fuel Particle Coating Failure Models and Data* [3], to guide the evaluation. NUREG-0111 addressed high enriched uranium UC₂ TRISO fissile particles with a 200 µm kernel and ThO₂ Binary Coated Particle (BISO) fertile particles with a 500 µm kernel, for service in a large prismatic HTGR. Major differences in particle design, fabrication specifications, and service conditions relative to the PBMR fuel, limit the applicability of this report to the PBMR fuel. Nonetheless, experience with this and other diverse fuel types over the course of High Temperature Reactor (HTR) fuel development has provided valuable insights into the development and understanding of the LEU UO₂ TRISO fuel.

2.3.4 Plans for Fuel Testing, Inspection and Surveillance

The SRP (page 4.2-20) states that for a fuel design that introduces new features, a more detailed surveillance program commensurate with the nature of the changes is warranted. The program should include appropriate qualitative and quantitative inspections to be carried out at interim and end-of-life refuelling outages. This surveillance program should be coordinated with prototype testing discussed in paragraph 2.3.3. When prototype testing cannot be performed, a special detailed surveillance program should be planned for the first irradiation of a new design. The MHTGR review was based on a preliminary safety

information document for a standardized design with a long-term objective of obtaining a final design approval and certification. The scope of the review did not include fuel testing, inspection, and surveillance. The online recirculation of PBMR fuel supports a practical approach to surveillance of fuel performance. A detailed program for examination and testing of selected irradiated fuel spheres following removal from the reactor is being planned for the initial PBMR power plant.

2.4 NRC PRECEDENTS INVOLVING LEU TRISO FUEL

Licensing interactions on Fort St. Vrain and large HTGR designs were based on HEU TRISO fuel, but have some general applicability to LEU fuel as recognized by the NRC staff in referencing NUREG-0111 [3] in the course of the MHTGR review. On two occasions during the last 25 years, the NRC has had occasion to consider LEU TRISO fuel for gas-cooled reactors. In December 1995, the NRC published the draft final results of its pre-application review of the LEU UCO TRISO fuelled MHTGR conceptual design, which began in January 1985, in NUREG-1338, *Preapplication Safety Report for the Modular High Temperature Gas-Cooled Reactor*. Additionally, in June 2002, the NRC staff provided Exelon with Requests for Additional Information (RAIs) related to the LEU UO₂ TRISO PBMR fuel [4] based on interactions begun in January 2001.

In the December 1995 draft final version of NUREG-1338, Section 4.2.1, the NRC staff noted that the MHTGR proposed use of high integrity fuel, with a very small defective fuel fraction for manufacturing, normal operations, and accidents. The NRC staff stated its belief 'that the fuel design and quality can be developed to meet the performance objectives.'

3. PBMR APPROACH

As established by decades of irradiation and testing over a wide range of particle designs [10], the behaviour of TRISO coated particle fuel is such that it does not experience widespread particle failure when a limiting parameter is exceeded. Instead, under simulated accident conditions at elevated temperature, a very small fraction of particle failures and/or fission product release are observed to occur gradually over a period of time (hours or days). During normal operation, particle failures are typically not observed, as the purpose of limiting the normal operation parameters is to assure acceptable performance under accident conditions, which is more restrictive than precluding failures during normal operation. There is no coolant phase change, no mechanisms for propagation of particle failures, and no autocatalytic reactions. Thus there is no distinct accident condition limit beyond which high releases would suddenly occur. Likewise, in normal operation, a short-term (hours) transient condition which exceeds the steady state operating temperature or power limits for long-term continuous operation will not significantly affect the state of the fuel.

The following paragraphs summarize the approach to achieving regulatory compliance for the PBMR fuel design:

- Paragraph 3.1 provides an overview of the PBMR reactor design and a comparison to German reactor designs.
- Paragraph 3.2 summarizes the PBMR fuel design and service conditions.
- Paragraph 3.3 describes the German design basis and operating experience.
- Paragraph 3.4 summarizes the PBMR fuel qualification program.

3.1 OVERVIEW OF PBMR REACTOR UNIT DESIGN

3.1.1 Design Description of PBMR Reactor Unit

The PBMR is a High Temperature Gas-cooled Reactor (HTGR) employing online fuelling and delivering 400 MW thermal power with a Reactor Outlet Temperature (ROT) of 900 °C. The purpose of the Reactor Unit (RU) is to provide heat that can be used in the Power Conversion Unit (PCU) to drive the turbine, which in turn converts thermal energy into mechanical shaft power to drive the generator. A cutaway schematic of the RU is shown in Figure 1. Schematic vertical and horizontal cross-sections through the reactor unit are shown in Figure 2 and Figure 3, respectively.

As shown in Figure 1, the RU consists of the fuel, the Core Structures (CS), the Reactivity Control System (RCS), and the Reserve Shutdown System (RSS). The equilibrium fuel core consists of approximately 452 000 fuel spheres in a loose annular pebble bed with an inner diameter of 2 m, an outer diameter of approximately 3.7 m, and an equivalent cylindrical height of approximately 11 m. In addition, approximately 30 000 fuel spheres are in the defueling region below the reactor core. The fuel spheres are inserted at the top of the core through three fuelling lines and are removed from the core at the bottom through three Core Unloading Devices (CUDs), shown in Figure 1. Under this scheme, each fuel sphere is recycled through the core an average of six times.

The fuel is based on German TRISO-coated fuel particles with Low Enriched Uranium (LEU) dioxide as developed for the HTR-Modul reactor. German LEU TRISO fuel was extensively used in the *Arbeitsgemeinschaft Versuchsreaktor* (AVR) and irradiated in three Materials Test Reactors (MTRs), all with satisfactory results. The coated particle is approximately 1 mm in diameter. Coated fuel particles are moulded into 50 mm diameter graphite spheres,

each containing approximately 14 500 coated particles. This 50 mm graphite sphere is then enclosed in a 5 mm thick, fuel-free graphite layer. The overall fuel sphere is then 60 mm in diameter. The fuel design and the German experience with this fuel are described in paragraphs 3.2 and 3.3. The fuel spheres are contained in a tall and slender annular core volume shaped and supported by the reflectors of the CS.

The main components of the CS are the metallic Core Barrel Assembly (CBA) and the Core Structure Ceramics (CSC). The CSC includes the reflectors that are grouped into the top, center, side, and bottom reflectors. The basic structural material of the reflectors is graphite, machined in the form of blocks. The graphite blocks are stacked in vertical columns which are supported by the CBA. The CBA is a steel cylindrical shell that is located and supported within the steel Reactor Pressure Vessel (RPV). The CSC allows the gas to flow through the pebble bed from the top to the bottom.

The Reactivity Control and Shutdown System provides two independent and diverse shutdown systems, namely the RCS and the RSS. The RCS consists of 24 control rod units containing B_4C that are divided into two control groups. The rods are suspended in the control rod channels located in the side reflector of the CSC. Control Rod Drive Mechanisms (CRDMs) raise and lower the control rods in the control rod channels. The CRDMs also release the control rods in the event of a reactor SCRAM, allowing them to drop, thereby adding the negative reactivity required to maintain the core at cold shutdown conditions. The primary rapid reactivity control response is from the prompt negative temperature coefficient, which shuts the reactor down passively as temperature rises.

The RSS consists of eight independent Small Absorber Sphere (SAS) insertion systems. The SAS are 10 mm diameter graphite spheres containing B_4C . The SAS are stored in containers in the RPV head. Core shutdown is achieved by releasing the absorber spheres that fall freely under gravity into the SAS channels located in the center reflector of the CSC. The absorber spheres are extracted from the SAS channels and transported back to the storage containers when they are no longer required.

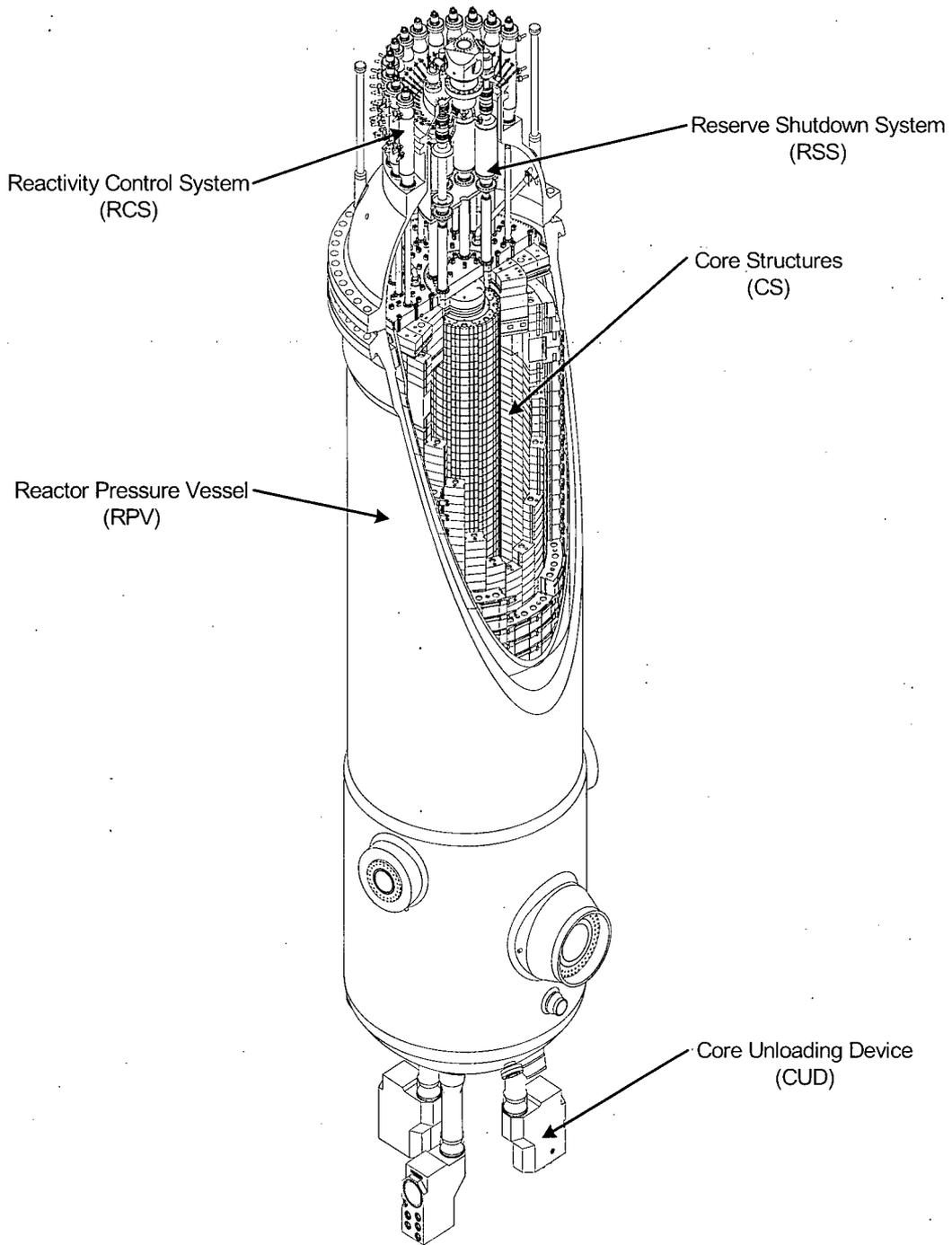


Figure 1: General Arrangement of the Reactor Unit inside the Pressure Vessel

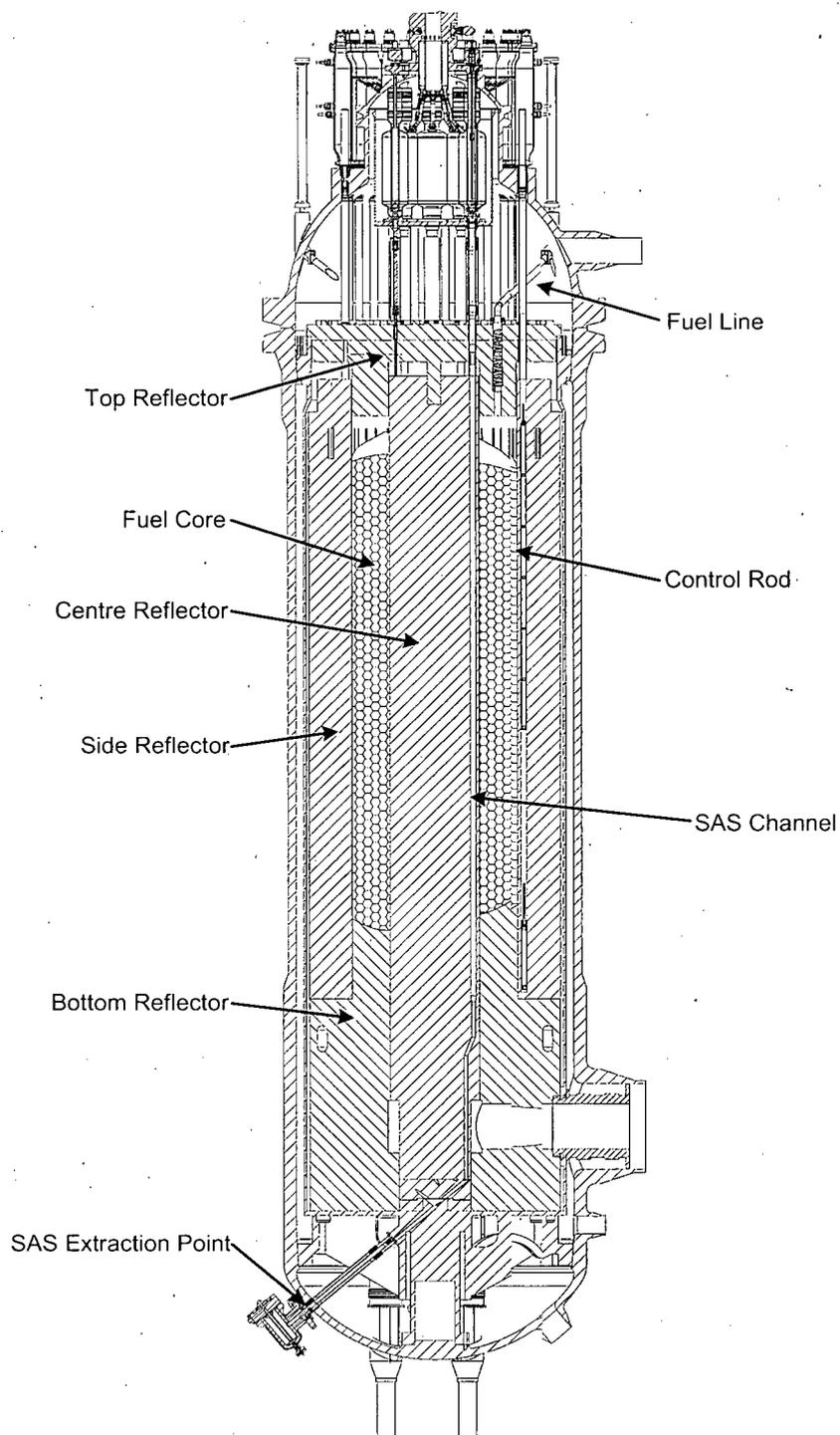


Figure 2: Vertical Schematic Section through the Pressure Vessel

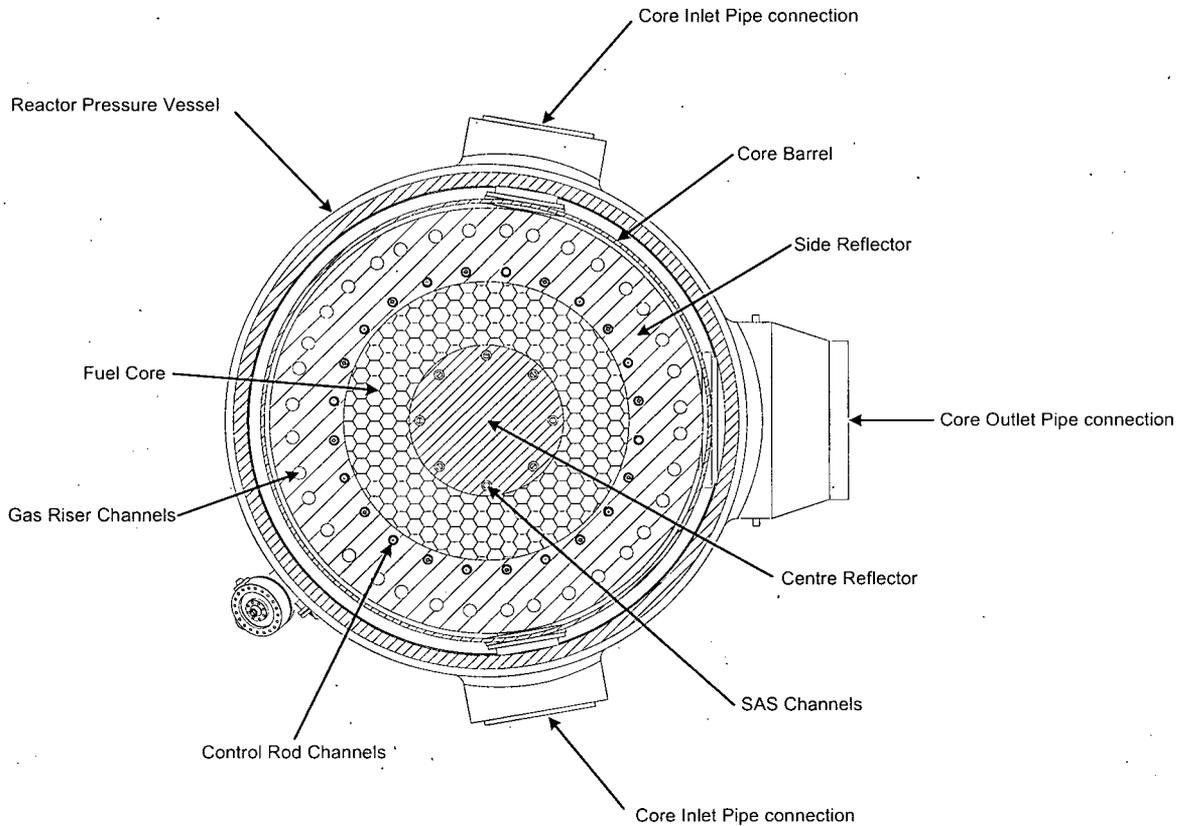


Figure 3: Horizontal Schematic Section through the Pressure Vessel

3.1.2 PBMR vs. German Reactor Design Comparison

PBMR fuel is based on the development, production, and operation of fuel for pebble bed reactors designed and built in Germany. The first pebble bed reactor to operate in Germany was the AVR and the second to operate was the THTR. Many of the design principles used in the PBMR RU are derived from THTR design principles. The HTR-Modul was a reactor concept, very similar to the PBMR, that was developed in Germany, with a tall, slender core in an uninsulated steel pressure vessel, and passive safety features for removal of heat from the reactor cavity.

Table 1 summarizes the principal mechanical, neutronic, gas, and thermodynamic design parameters of the PBMR RU compared to other pebble bed reactor designs in Germany.

Table 1: Reactor Unit Comparison

Parameter	PBMR	HTR-Modul	THTR	AVR
Reactor thermal power	400 MW	200 MW	750 MW	46 MW
Heat transport medium	Helium	Helium	Helium	Helium
Reactor inlet temperature	500 °C	250 °C	250 °C	275 °C
Reactor outlet temperature	900 °C	700 °C	750 °C	950 °C
Mass flow rate	192 kg/s	85 kg/s	296 kg/s	13 kg/s
System operating pressure	9 MPa	6 MPa	3.9 MPa	1.08 MPa
Pressure Vessel	Steel	Steel	Pre-stressed concrete	Steel
Reactivity Control System	24 control rods in the side reflector	6 control rods in the side reflector	36 control rods in the side reflector	4 control rods in side reflector buttresses
Reserve Shutdown System	8 channels in the center reflector filled with absorber spheres	18 channels in the side reflector filled with absorber spheres	42 rods inserted into the pebble bed	
Fuel type	Spherical	Spherical	Spherical	Spherical
Fuel sphere diameter	60 mm	60 mm	60 mm	60 mm
Fuelling method	Multiple recycle	Multiple recycle	Multiple recycle	Multiple recycle
Helium flow direction	Downwards	Downwards	Downwards	Upwards
Pebble bed inner diameter	2.0 m	Not applicable	Not applicable	Not applicable
Pebble bed outer diameter	3.7 m	3.0 m	5.60 m	3.0 m
Pebble bed height	11.0 m	9.4 m	6.00 m	2.8 m
Volume of pebble bed	~84 m ³	~66 m ³	~125 m ³	~20 m ³
Number of fuel spheres	~452 000	~360 000	~675 000	~100 000
Mean power density	4.8 MW/m ³	3.0 MW/m ³	6 MW/m ³	2.6 MW/m ³
Mean fuel sphere output	~0.9 kW	~0.6 kW	~1.1 kW	~0.5 kW
Status	Being designed	Cancelled	Operated	Operated

3.2 PBMR FUEL DESIGN DESCRIPTION AND SERVICE CONDITIONS

The core consists of fuel elements, containing uranium, that generate heat by means of fission reactions. The fuel element, known as a fuel sphere, consists of a matrix graphite body pressed into a spherical shape. A fuel sphere is divided into two regions. The inner spherical region is known as the fuel region, while the outer shell surrounding the fuel region is known as the fuel-free region. The fuel region of each fuel sphere contains a large number of evenly dispersed spherical particles, known as coated particles, in which the fuel is contained, while there are no coated particles in the fuel-free region.

The design of the coated particles and fuel sphere is depicted in Figure 4.

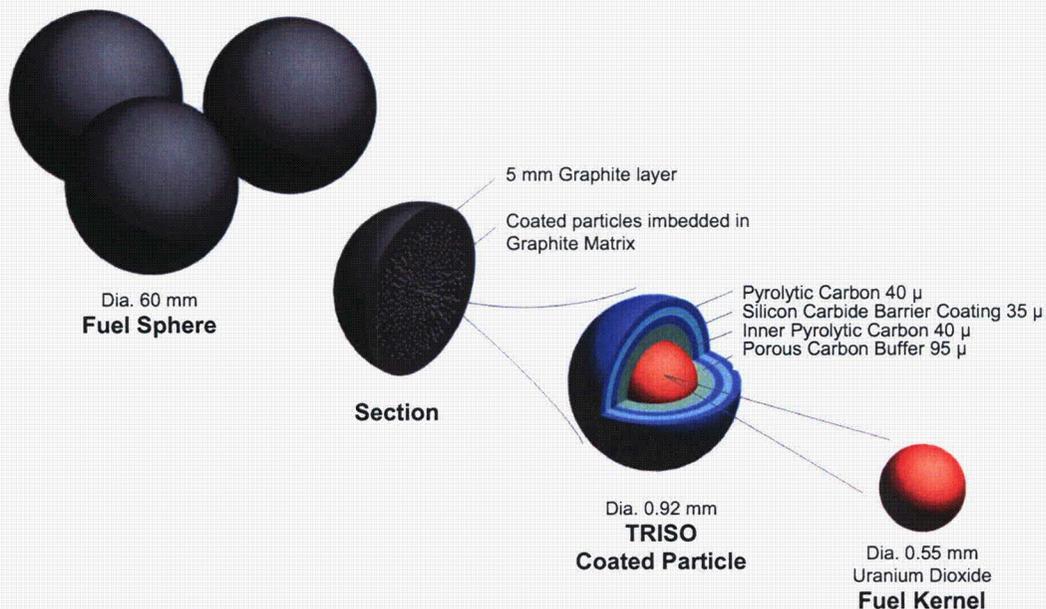


Figure 4: Fuel Sphere Design

3.2.1 Coated Particle Design

The properties of LEU-TRISO coated particles are among the most important factors determining the radiological safety of the PBMR. This is because fission product retention in fuel spheres, and the fuel burnups and temperatures that can be tolerated in the reactor core, are primarily determined by coated particle properties.

3.2.1.1 Kernel

The spherical fuel kernel consists of stoichiometric uranium dioxide (UO_2). The kernel serves as an important barrier to radionuclide release by immobilizing many of the fission products and delaying the diffusive release of others, substantially reducing release from the particle due to radioactive decay before release from the kernel.

The basic manufacturing steps for the kernel are as follows:

- U_3O_8 powder is dissolved in nitric acid to form uranyl nitrate.
- The solution is neutralized with ammonia and allowed to flow through an oscillating nozzle to produce spherical droplets.
- As the droplets fall through a gaseous ammonia atmosphere, the spherical outer surface of the droplet gels.
- The particles fall into an aqueous ammonia solution, where they solidify into ammonium uranate.
- The particles are then aged and washed to remove ammonium nitrate and organic additives, dried, and calcined.
- The dry kernels are reduced to UO_2 in hydrogen and sintered, and are then ready to be coated.

3.2.1.2 Buffer layer

The first layer in contact with the kernel is known as the buffer layer; it is deposited from acetylene (C_2H_2) in a heated fluidized-bed. The temperature and other conditions in the fluidized bed are adjusted to produce a porous pyrocarbon layer that has approximately 50% of the theoretical density of pyrocarbon.

The purpose of the buffer layer is to provide void volume for gaseous fission products in order to limit pressure buildup within the coated particle. It also serves to decouple the kernel from the inner pyrocarbon layer to accommodate kernel swelling, thereby reducing the buildup of stress in the outer coating layers during irradiation. The buffer layer also absorbs energetic fission products recoiling from the kernel surface, thus protecting the inner pyrocarbon layers of the coated particle.

3.2.1.3 Inner pyrocarbon layer

The inner high-density, isotropic layer of pyrolytic carbon is also referred to as the Inner Low Temperature Isotropic (ILTI) pyrocarbon layer. It is deposited from a mixture of acetylene and propylene in a heated fluidized-bed, and has an average density of approximately 1.9 g/cm^3 .

The ILTI layer forms the first load-bearing barrier against the pressure exerted by fission products within the fuel kernel and buffer layer, thereby reducing the pressure on the next layer, which consists of silicon carbide (SiC). During irradiation, the ILTI and Outer Low Temperature Isotropic (OLTI) layers shrink at first, possibly expanding again if sufficiently high fast neutron dose levels are reached. The interaction between the ILTI and OLTI high-density pyrocarbon layers and the SiC layer sandwiched between them plays an important role in keeping the SiC layer under compressive stress as long as possible during irradiation.

Although an intact ILTI layer forms a practically impenetrable barrier for fission gases and iodine, it becomes increasingly pervious to cesium, silver, and strontium at higher temperatures.

3.2.1.4 Silicon carbide layer

When SiC is deposited from methyltrichlorosilane under the correct conditions, a density of $\sim 3.2 \text{ g/cm}^3$ (nearly 100% theoretical density) with a grain size that is small in comparison to

the SiC layer thickness is obtained. The resulting SiC layer has high strength and low permeability.

As noted above, at high temperatures, the ILTI and OLTl layers partially lose their ability to contain cesium, silver, and strontium. A primary purpose of the SiC layer is to prevent the release of these fission products into the graphite matrix, and then into the reactor helium stream. The SiC layer thus acts as the principal metallic fission product retention barrier in the coated particle. The SiC layer also has sufficient strength to withstand internal pressure produced during irradiation (e.g., the PANAMA code [5] particle failure model neglects the beneficial effects of the ILTI and OLTl layers). However, the coated particle structure, in conjunction with the irradiation induced shrinkage of the ILTI and OLTl, results in the SiC layer being kept under compression for most of the irradiation. This provides a high level of assurance that the SiC layer will remain intact under all foreseeable reactor core conditions.

3.2.1.5 Outer pyrocarbon layer

The OLTl pyrolytic carbon layer is deposited in the same way as the ILTI layer. The function of this layer is to protect the SiC layer against damage in the fuel manufacturing processes following the coating process. It also provides pre-stress on the outside of the SiC layer, due to its net shrinkage under fast neutron irradiation during the fuel lifetime in the reactor core, thereby reducing the tensile stress in the SiC layer, and serves as a redundant barrier to gaseous fission product release.

During isostatic pressing, un-round coated particles show a much greater tendency to crack than spherical particles. Therefore, wet chemical processes that produce highly spherical kernels are used during the initial stages of kernel manufacture. Also, coating process conditions that produce coatings of uniform thickness are established and maintained. Coated particle failure in fuel spheres can be greatly reduced by removing particles that show an unacceptable deviation from a spherical shape. Unacceptably out-of-round particles are removed on a sorting table whose surface is slightly inclined in one direction as it is vibrated.

3.2.1.6 Overcoating

In preparation for fuel sphere manufacturing, a coating of finely ground matrix graphite and resin binder is applied to the outer surface of each coated particle in a rotating drum. This coating is known as the 'overcoat.' Its purpose is to prevent coated particles from coming into contact with each other, thereby damaging their coatings during pressing of the fuel spheres.

3.2.2 Fuel Sphere Design

A fuel sphere is formed by pressing coated particles overcoated with matrix material, along with additional matrix material, into a 50 mm inner sphere, each of which contains approximately 14 500 coated particles. The inner fuel sphere is then placed within a protective 5 mm thick layer of matrix material formed by an isostatic pressing process, and machined to final dimensions.

3.2.2.1 Matrix graphite

The function of the matrix graphite is to contain the coated particles in a fuel sphere, protect them from mechanical damage, and provide a heat conduction path between the coated particles and the reactor coolant, helium. The carbon in the matrix also acts as the moderator

for neutrons in the PBMR core. Highly graphitized (i.e., aligned crystalline structure) materials are used for matrix graphite for the following reasons:

- Highly graphitized material ensures dimensional stability during irradiation with fast neutrons, as partially graphitized material will undergo further graphitization under fast neutron irradiation, with accompanying dimensional changes.
- Once a fuel sphere has been pressed, it is no longer possible to change the degree of graphitization of the materials contained in the fuel sphere. The reason is that temperatures required for graphitization (2 700 °C to 3 000 °C) would also damage the effectiveness of the SiC layer.
- Highly graphitized material has the desirable property that it can be relatively easily pressed to the required density.

3.2.2.2 Pressing

Fuel spheres are pressed at high pressure, without application of external heat, to obtain the required density that ensures adequate structural stability and heat conduction. This also provides the correct amount of carbon in the reactor core to determine heat capacity and moderation.

PBMR fuel spheres are pressed in two steps:

- In the first step, coated particles and matrix material are mixed and pressed to form a fuel-containing region consisting of a sphere 50 mm in diameter. Coated particles are distributed evenly in this inner fuel-containing region to prevent the development of hot spots in the fuel sphere.
- In the second step, matrix material is added to the mould and pressed to form a 5 mm thick fuel-free region around the fuel-containing region. The purpose of this region is to protect the inner zone from mechanical and chemical damage during handling and operation.

Final heat treatment steps remove organic components and impurities and make the spherical fuel strong and corrosion resistant.

3.2.3 PBMR Fuel Service Conditions

Experimental results and analytical modelling of LEU UO₂ TRISO fuel have indicated that the integrity of the coated particles and retention of fission products can be assured if the fuel is properly manufactured (i.e., according to the manufacturing specifications), and the normal operation and accident service conditions are maintained within an allowed performance envelope. This section addresses the range of service conditions expected for PBMR fuel, which replicates German LEU UO₂ TRISO fuel. The fuel temperature, power, and flux levels are calculated using the VSOP99 code [6]. The existing experimental basis for establishing an allowed envelope for the German LEU UO₂ TRISO fuel is addressed in paragraph 3.3. The parameters which influence PBMR fuel performance are the temperature history, burnup and fast fluence experienced during normal operation, and the temperature history experienced during accident conditions. The temperature gradient, in combination with the other parameters, has been seen to drive kernel migration, particularly in fixed fuel configurations where the gradient is essentially constant for long periods. As discussed on page 46 of [7], this phenomenon has been observed in and is a concern for prismatic designs, which have higher fuelled region (compacts) power densities and corresponding higher radial heat flux, with some of the fuel remaining static in a high power location for long

periods. It is effectively controlled in the PBMR by limiting the maximum sphere power and the constant movement resulting from continuous cycling of fuel spheres. The other parameters and their expected values for the PBMR are discussed in the following paragraphs and summarized in paragraph 3.2.3.5.

3.2.3.1 Fuel temperature, normal operation

The chemical interaction and diffusion processes that can affect coated particle integrity and fission product retention are governed by time and temperature, with the rate constants typically following an exponential Arrhenius function of temperature. Thus the integrated temperature history, not the instantaneous temperature value, is of primary interest during normal operation. Fuel spheres are inserted onto the top of the PBMR core from three symmetric positions around the core annulus. Since both coolant and sphere flow are downward, the initial fuel temperature is only slightly greater than the inlet coolant temperature, with temperatures increasing as the spheres flow through the core. The core is modelled by five annular flow channels, tracking numerous layers within each of the channels as they pass downward through the core. Fuel temperatures in the PBMR vary as the spheres move through the core, with equilibrium core temperature distributions as shown in Figure 5. The channels listed in Figure 5 are annular core segments, with channel 1 being the segment next to the inner reflector, channels 2 through 4 interior channels, and channel 5 next to the outer reflector. The temperatures shown are sphere center temperatures based on the average power in the layer, which contains a mixture of spheres on different passes through the core. The highest temperature (1 106 °C) is calculated for a sphere in Channel 1 on its first pass at slightly below the core midplane. The core average temperature, which corresponds to the time-averaged temperature experienced by the average fuel sphere, is 904 °C.

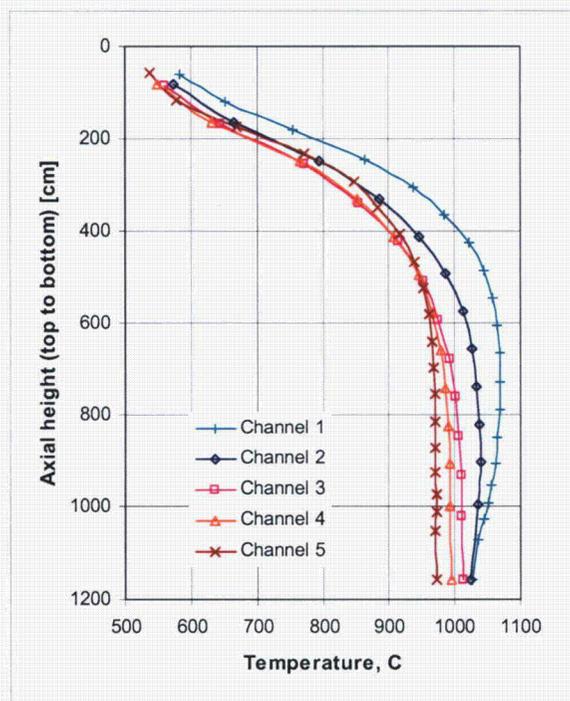


Figure 5: PBMR Normal Operation Fuel Temperature (Axial Profiles for Average Values in VSOP99 Channels 1 through 5)

3.2.3.2 Fuel burnup, normal operation

On average, fuel spheres pass through the PBMR core six times before exceeding the discharge setpoint of the burnup management system. The sphere power and corresponding burnup rate vary as a function of position within the core and recirculation pass. Figure 6 shows the axial power distribution for the average of the spheres in each layer of the five flow channels. The sphere powers within each layer vary according to the recirculation pass, with low burnup spheres on initial or early passes having higher power, and higher burnup spheres on later passes having lower power. Spheres passing near the inner reflector (Channel 1) have higher power due to the higher thermal flux. The volume averaged fuel burnup in the equilibrium core is calculated to be 5.5% FIMA. The discharge burnup distribution is primarily determined by the burnup management system setpoint, above which the spheres are routed to spent fuel storage. The setpoint is not firmly established at present, but initial studies indicate an average discharge burnup of ~9.5% FIMA, with a maximum of ~10.5% FIMA (~11.2% FIMA including an error allowance on the burnup measurement). A small number of spheres would approach the maximum discharge burnup, as they would have to have a multi-pass burnup just below the setpoint including error, experience a large negative measurement error, and be recirculated near the inner reflector (i.e., Channel 1).

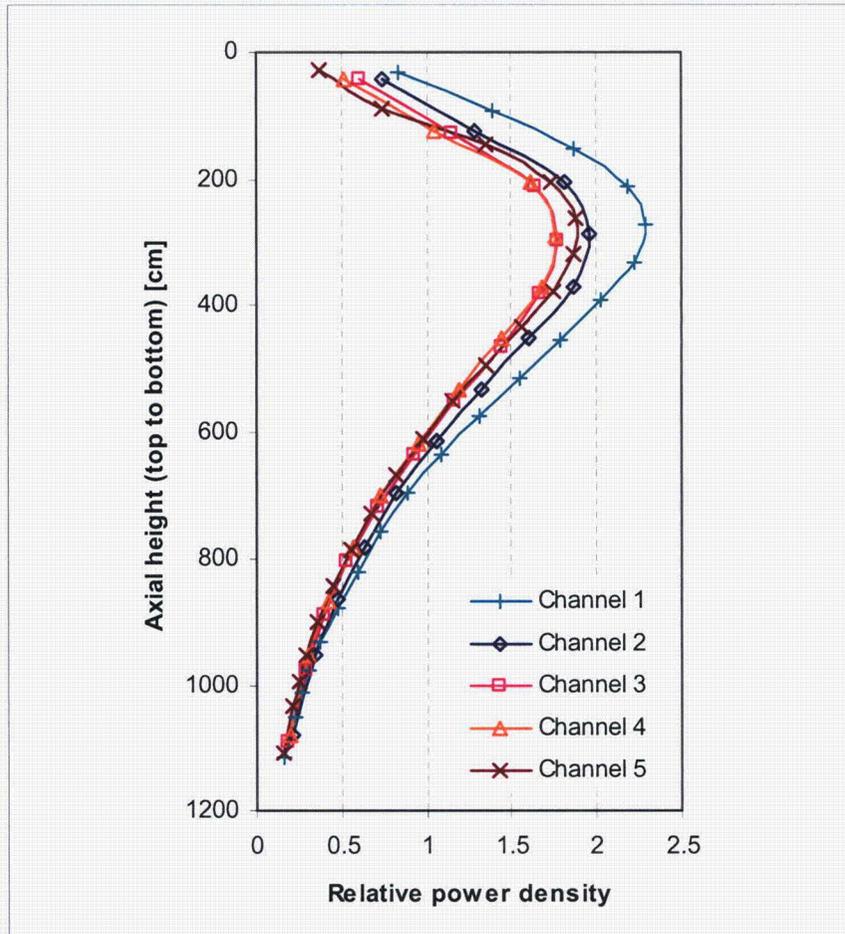


Figure 6: PBMR Normal Operation Power Distribution

3.2.3.3 Fast fluence, normal operation

On average, there is a one-to-one relationship between burnup and fast fluence for PBMR fuel, but the two parameters can vary independently for individual spheres due to the different spatial dependence of thermal flux (related to power and burnup accrual) and fast flux, as illustrated in Figure 7. In Figure 7, the thermal flux peaks near the reflectors while the fast flux peaks near the center of the annulus.

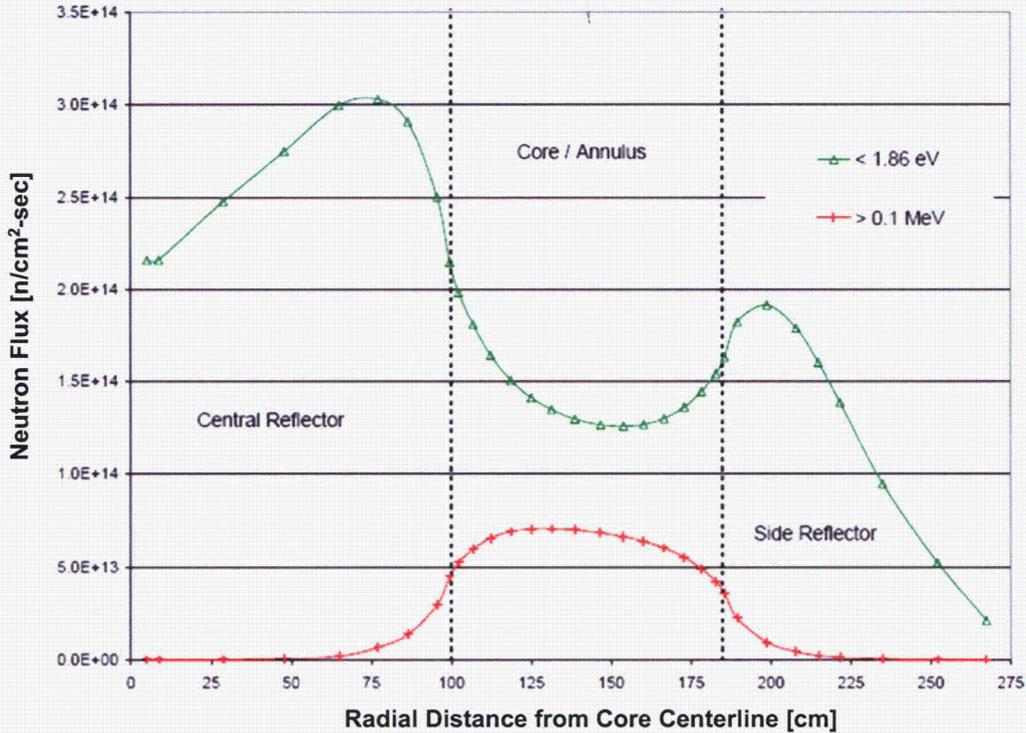


Figure 7: Radial Neutron Flux Distribution

A sphere passing near the reflector will accrue relatively higher burnup and lower fast fluence than a sphere passing in the interior of the annulus. Thus the ratio of burnup to fast fluence of individual spheres is dependent on their trajectories through the core. The net result is a distribution of burnup and fast fluence over a range, as illustrated in Figure 8. The VSOP99 analyses produced an equilibrium core average fast fluence of $\sim 1.37 \times 10^{21}$ n/cm² with an average discharge fluence of $\sim 2.65 \times 10^{21}$ n/cm². Initial estimates of maximum discharge fast fluence based on operation of the burnup management system indicate a maximum value of $\sim 3.2 \times 10^{21}$ n/cm² ($\sim 3.6 \times 10^{21}$ n/cm² including an error allowance on the burnup measurement). As was the case with burnup, only an extremely small fraction of discharged spheres would approach the maximum value including error allowance.

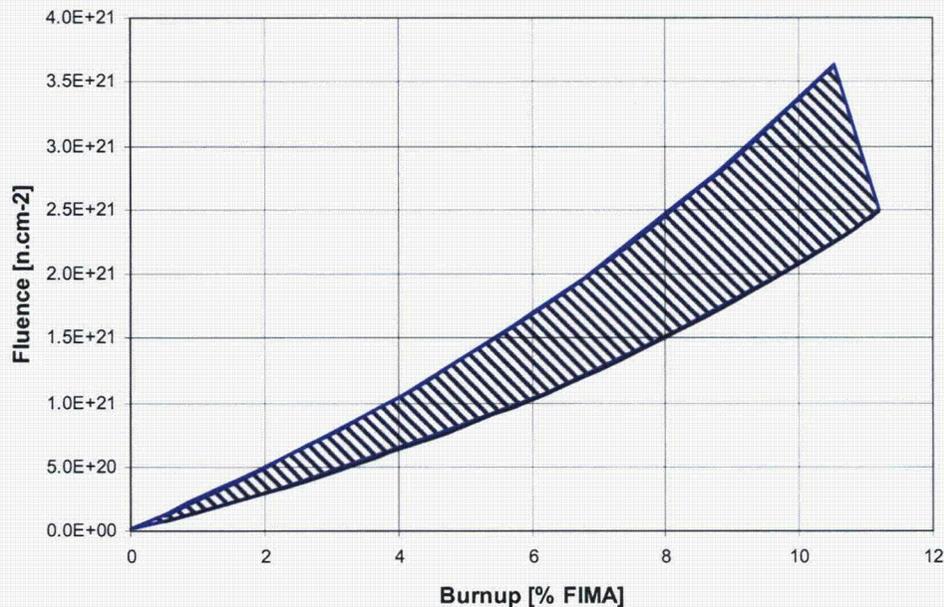


Figure 8: PBMR Burnup and Fluence Envelope

Within the envelope shown in Figure 8, the population would be skewed toward the upper side, as most of the sphere passes are through the center portion of the core annulus. Few spheres would be present in the region above 10% FIMA.

3.2.3.4 Fuel temperature, accident conditions

The TINTE (Time Dependent Neutronics and Temperatures) code was developed by the KFA Forschungszentrum Jülich to investigate nuclear and thermal transient behaviour of HTRs, with full neutron, temperature, and xenon feedback effects taken into account in two-dimensional r-z diffusion theory. TINTE was developed and has been benchmarked against AVR data [8] and test facility data [9]. Due to the large heat capacity of graphite, overall temperature changes are generally very slow in PBMR fuel. In contrast, any changes in the UO_2 kernel temperature will result in an immediate feedback to the nuclear behaviour. The code is therefore able to handle feedback effects over a very large range of time constants. The code incorporates numerous material property correlations for graphite and other core structures, including the temperature and fast-fluence dependence of the pebble bed effective thermal conductivity.

The Pressurized Loss of Forced Cooling (PLOFC) transient is an anticipated operational occurrence. A typical best-estimate maximum fuel temperature for this event is 1 319 °C. The Depressurized Loss of Forced Cooling (DLOFC) event is an accident for which a typical best-estimate maximum fuel temperature is 1 593 °C. Typical time-dependent traces of the maximum and average fuel sphere temperatures for the DLOFC accident are shown in Figure 9. From this figure it can be seen that relative to the total duration of the transient (> 100 h), the maximum temperatures are experienced for only a few (~10) hours.

TINTE analysis also shows that for the typical DLOFC accident, only small fractions of the spheres in the core approach temperatures which could result in prediction of failure of a few coated particles within the sphere. Figure 10 is a histogram of DLOFC fuel temperatures at the time of the maximum temperature. This figure shows that approximately 6.9% of the core

would experience temperatures in the range of 1 500 °C to 1 600 °C. Further, the best-estimate maximum temperature of 1 593 °C is reached by only ~0.35% of the spheres in the core. Figure 11 illustrates the evolution of the temperature distribution during the transient, indicating that a small fraction of the core exceeds 1 500 °C and that the highest temperature fuel is above 1 500 °C for less than 80 h.

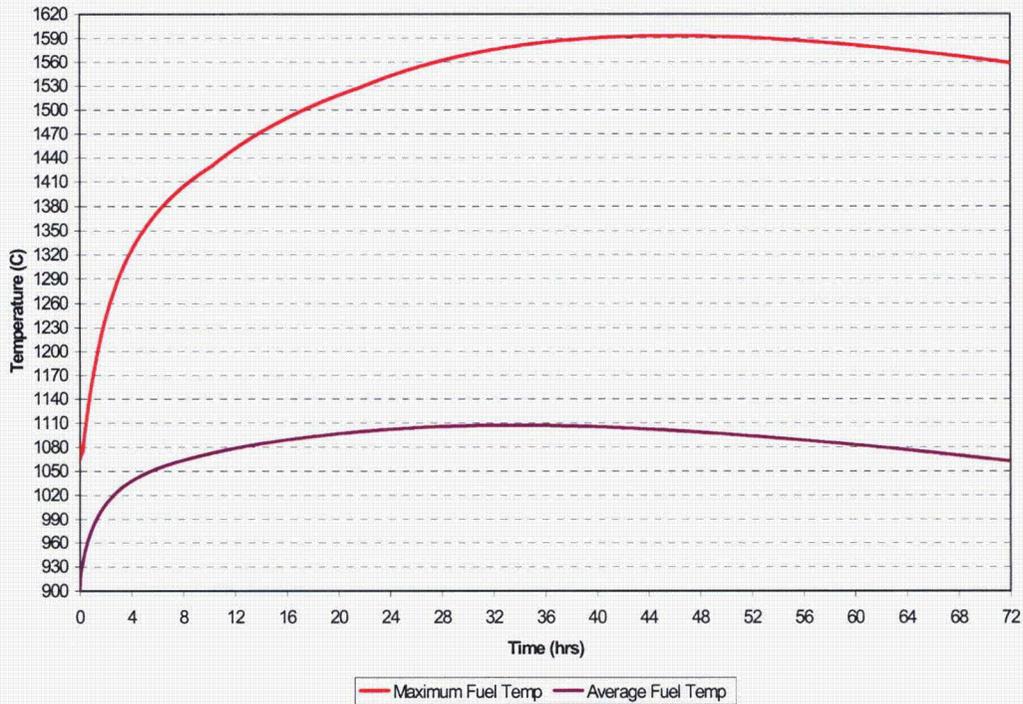


Figure 9: Fuel Temperature – Depressurized Loss of Forced Cooling

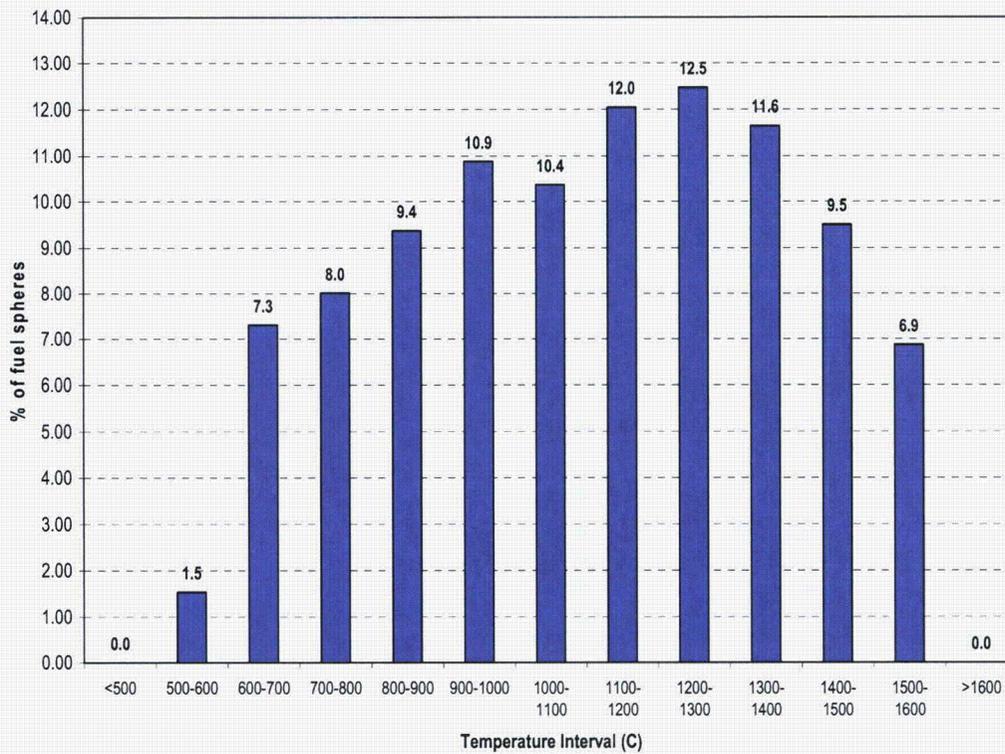


Figure 10: Fuel Temperature Distribution at Time of Maximum Temperature – DLOFC

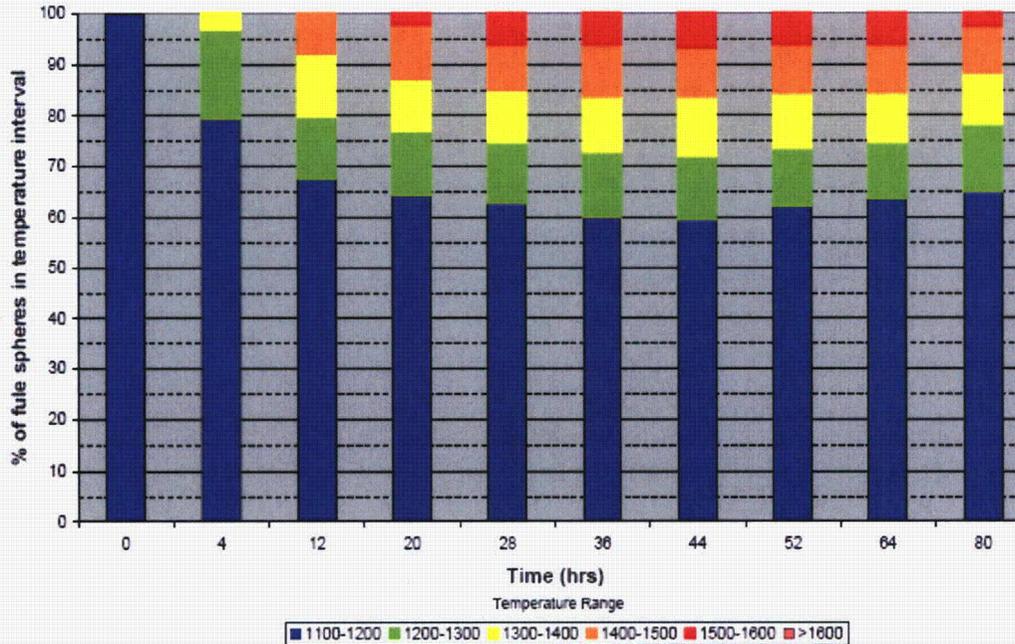


Figure 11: Fuel Temperature Distribution during Transient – DLOFC

3.2.3.5 Summary of PBMR fuel service conditions

Based on the data discussed in the previous paragraphs, the PBMR fuel service conditions are summarized in Table 2. These values will be used to compare with the existing data in paragraph 3.3.

Table 2: Fuel Service Conditions

Parameter	Core Average Value	Maximum Value
Normal Operation		
Burnup, % FIMA	5.5	11.2
Fast Fluence, 10^{21} n/cm ² (E > 0.1 MeV)	1.4	3.6
Sphere Center Temperature, °C	904	1 106
Accident Conditions – Maximum Transient Conditions		
Sphere Temperature, °C	1 100	1 593

3.3 GERMAN FUEL DESIGN AND OPERATING EXPERIENCE

As noted earlier, the basis for the PBMR fuel is the German LEU UO₂ TRISO fuel design and test experience. The development of this design drew from an extensive international background of coated particle fuel fabrication and testing experience that covered a broad range of parameters as summarized here:

- a. Kernel characteristics
 - Fissile/fertile materials – uranium, thorium, plutonium (mixed and unmixed).
 - Chemical forms – oxide, carbide, oxycarbide.
 - Enrichment – ranging from natural uranium to high enriched uranium and plutonium.
- b. Coating characteristics
 - BISO – variations in buffer and dense pyrocarbon coating thicknesses and properties.
 - TRISO – variations in buffer, dense pyrocarbon and silicon carbide (or zirconium carbide) thicknesses and properties.
- c. Fuel forms
 - Spheres – multiple geometries and fabrication methods.
 - Compacts – cylindrical and annular shapes with wide variations in particle packing fractions.
- d. Irradiation facilities
 - Materials Test Reactors – HFR (Netherlands), FRJ 2 DIDO (Germany), IVV-2M (Russian), Siloe (France), R2 (Sweden), BR2 (Belgium), HFIR (United States), ATR (United States), with wide variations in spectra and degree of acceleration.
 - Research and Demonstration Reactors – DRAGON (United Kingdom), Peach Bottom I (United States), AVR (Germany), Fort St. Vrain (United States), THTR (Germany).
- e. Irradiation and testing conditions
 - Burnup – ranging from < 1% to > 70% FIMA.
 - Fluence – ranging from < 1×10^{21} to > 10×10^{21} n/cm².

- Irradiation Temperature – ranging from 600 °C to 1 500 °C.
- Accident Simulation Temperature – ranging from 1 400 °C to 2 500 °C.

This broad range of experience and data has supported development of an understanding of the parameters and phenomena of importance in the fabrication and performance of coated particle fuel. Extensive bilateral and multilateral international information exchanges facilitated the incorporation of this broad experience base into the German and other modern coated particle fuels. A detailed review of the US and German experience and its relationship to fuel performance and fuel performance modelling is documented in an Electric Power Research Institute (EPRI) report [10]. The evolution of the German fuel design, arriving at the LEU UO₂ TRISO pressed sphere selected as a basis for the PBMR, is summarized in Section 2.4 of a report on the AVR [11]. In considering this experience and data, the international community has converged on a common LEU TRISO particle design with very similar coating thicknesses and properties, with variations in kernel diameter, enrichment, and composition (UO₂ and UCO), depending on the specific service conditions and requirements. The German LEU UO₂ TRISO particle design that is the basis for the PBMR fuel is described in paragraph 3.3.1, with data from [12].

3.3.1 Design and Manufacturing Experience

Fuel spheres produced in Germany for the HTR-Modul reactor Proof Tests were adopted as the reference design for PBMR fuel, with small adjustments in enrichment and sphere particle loading to address PBMR service conditions. These Proof Test fuel spheres were produced in Germany in 1988, and represent the culmination of the German LEU-TRISO fuel development and qualification program conducted in the period from 1981 to 1988. A large body of experimental data obtained by means of an irradiation and PIE program, covering a wide range of operating parameters, supports the German LEU UO₂ TRISO fuel design. This database supports establishment of an operating envelope for this fuel design, covering normal operation as well as DBA conditions.

In pebble fuel, coated particles provide the main barrier against release of fission products, thus attention is primarily focused on performance of the coated particles. Although the fuel sphere provides additional fission product retention through diffusion and trapping and adsorption effects, the principal function of the sphere is to protect the embedded coated particles against external environmental and mechanical effects, and to facilitate fuel handling. This means that all irradiation test results obtained on fuel samples containing coated particles of a design similar to that for PBMR can be included in the experience database when considering particle performance.

The LEU-TRISO fuel types manufactured and tested in Germany are summarized in Table 3. Fuel spheres intended for AVR operation were manufactured in large numbers for the purpose of bulk testing in a reactor environment. Fuel spheres manufactured for the German LEU Phase 1 irradiation test program and for the Proof Test for the HTR-Modul were manufactured in smaller numbers.

Table 3: LEU UO₂ TRISO Fuels Manufactured and Tested

Characteristic	Pre-1985 Production			Post-1985 Production	
	1981	1981	1983	1985	1988
Year of Manufacture	1981	1981	1983	1985	1988
Designation	GLE 3	LEU Phase 1	GLE 4	GLE 4/2	Proof Test Phase 2
Matrix Material	A3-27	A3-27	A3-27	A3-3	A3-3
Irradiation Test Designation	AVR 19	HFR-K3 FRJ2-K13 HFR-P4 SL-P1 FRJ2-P27	AVR 21-1 FRJ2-K15	AVR 21-2	HFR-K5 HFR-K6
Approximate number of fuel spheres manufactured	24 600	100	20 500	14 000	200

Notes:

The symbols used in the 'Irradiation Test Designation' row have the following meanings:

- The first three letters describe the reactor in which the test was done:
 - AVR = *Arbeitsgemeinschaft Versuchsreaktor* in Jülich, Germany
 - HFR = High Flux Reactor in Petten
 - FRJ2 = DIDO reactor in Jülich
 - SL = Siloe reactor in Grenoble
- The next group of symbols describe the irradiation sample type and test number. In the case of AVR irradiations, the reload number is used, i.e., AVR 19 means that the fuel spheres made up the 19th partial reload of the reactor. In other tests, the letter K designates a full-sized fuel sphere, the letter P designates coated particles in any other form, i.e., small spheres, compacts or coupons, and the number is the test number. Thus FRJ2-P27 means irradiation test number 27 performed on coated particles in the DIDO reactor in Jülich.

Manufacturing detail for the different LEU-TRISO fuel types of Table 3 is given in Table 4.

Table 4: Manufacturing Detail for LEU-TRISO Fuel Types

Characteristic	Pre-1985 Production			Post-1985 Production		PBMR Specification
	GLE 3	LEU Phase 1	GLE 4	GLE 4-2	Proof Test Phase 2	
Kernel Diameter (μm)	500	497	501	502	508	500
Kernel Density ($\text{g}\cdot\text{cm}^{-3}$)	10.80	10.81	10.85	10.87	10.72	> 10.4
Coating Thickness (μm)						
• Buffer Layer	93	94	92	92	102	95
• Inner PyC Layer	38	41	38	40	39	40
• SiC Layer	35	36	33	35	36	35
• Outer PyC Layer	40	40	41	40	38	40
Coating Density ($\text{g}\cdot\text{cm}^{-3}$)						
• Buffer Layer	1.01	1.00	1.01	1.1	1.02	< 1.05
• Inner PyC Layer	1.86	~1.9	1.9	1.9	1.92	1.9
• SiC Layer	3.19	3.20	3.20	3.2	3.20	≥ 3.18
• Outer PyC Layer	1.89	1.88	1.88	1.9	1.92	1.9

Characteristic	Pre-1985 Production			Post-1985 Production		PBMR Specification
	GLE 3	LEU Phase 1	GLE 4	GLE 4-2	Proof Test Phase 2	
Loading						
• Heavy Metal (g/FS)	10	10	6	6	9.4	9
• Uranium 235 (g/FS)	1	1	1	1	1	0.86
• Enrichment (% U-235)	9.82	9.82	16.76	16.76	10.6	9.6
• Coated Particle per FS	16 400	16 400	9 560	9 560	14 580	14 440
Free Uranium Fraction ($\times 10^{-6}$)	50.7	35	43.2	7.8	13.5	< 60

While German manufacturing experience dates from the 1960s, it is the LEU UO_2 TRISO fuel manufacturing, which started in 1981, that is directly applicable to PBMR. Table 4 shows that the pre-1985 and post-1985 fuel designs are nearly identical, except for enrichment and heavy metal loading in the spheres. Although the enrichment and heavy metal loading varied, the amount of ^{235}U per sphere was kept constant at approximately 1 g.

The delineation between pre-1985 and post-1985 is not based on the fuel design, but rather on two particular improvements in the manufacturing process. Coated particles are 'overcoated' with graphite prior to mixing them with matrix graphite in preparation for pressing of the fuel sphere. For the pre-1985 category, the overcoating of the particles was done manually, whereas for the post-1985 category, overcoating was automated using a specially designed mixer operated by a robot. This change in the overcoating process and the introduction of vibration tables in three stages to remove aspherical kernels, particles, and overcoated particles during particle manufacturing, resulted in a significant improvement in the 'free uranium' burn-leach test results for completed fuel spheres. The free uranium fraction decreased by about a factor of four from the average of the pre-1985 results to the average of the post-1985 results (refer to Table 4).

Since the manufacturing process change that delineates the two categories significantly impacts the determination of the free uranium fraction, only the burn-leach test results from the post-1985 category were used in the calculation of the expected 'coated particle failure fraction' due to manufacturing defects. Forty GLE 4/2 spheres and 10 Proof Test spheres (Table 3) containing a total of 528 200 coated particles were subjected to the burn-leach test. Test results determined that the free uranium in these 528 200 particles was equivalent to the uranium in six coated particles. Therefore, the sample mean failure fraction is 1.1×10^{-5} and the expected failure fraction (50% confidence that population fraction is no higher) due to manufacturing is 1.3×10^{-5} .

The substantial majority of the fuel irradiation and testing data discussed in paragraph 3.3.2 was produced from the GLE 3 and LEU Phase I fuels. As shown in Table 4, the parameters for these fuels are nearly identical to PBMR fuel. The primary areas of difference are a somewhat higher number of particles per sphere, and a higher free uranium fraction (relative to the expected value for PBMR). These differences are conservative with respect to use of the data in the context of expected performance of PBMR fuel.

3.3.2 Normal Operation Irradiation Performance

German LEU UO_2 TRISO fuel normal operation irradiation experience can be classified into two categories: large-scale irradiation in an operating reactor and fuel specimen irradiation in MTRs.

- The AVR Test Reactor program for LEU UO_2 TRISO fuel covered bulk testing of fuel spheres under actual operating conditions in a pebble bed core. Mass-produced fuel

spheres were irradiated in AVR to design burnup, with some of the fuel spheres removed from AVR at different burnup stages and subjected to ex-reactor heating tests to simulate performance under accident heatup conditions. Both GLE 3 and GLE 4 spheres were irradiated, but heating tests focused on GLE 3 spheres. The results of the heating tests are discussed in paragraph 3.3.3, but the initial release from the heatup testing was used to infer the state of the fuel spheres following irradiation (i.e., normal operation performance).

- MTR irradiations were conducted in three reactors: HFR (Netherlands), FRJ2 (Germany) and Siloe (France), and in two phases:
 - Phase 1 included normal fuel spheres, small fuel spheres, compacts, and coupons containing coated particles from the same coating batch (EUO 2308) irradiated under a variety of conditions. These irradiation tests were intended to provide an operation envelope for a number of different reactor types designed around LEU UO₂ TRISO fuel. One additional irradiation of a GLE 4 fuel sphere was conducted in DIDO (FRJ2-K15).
 - Phase 2 was performed to provide MTR irradiation proof-tests specifically for the HTR-Modul reactor design. During Phase 2 tests, fuel spheres were subjected to temperature cycles derived from those expected during normal operation of the HTR-Modul. The German fuel program was terminated before the irradiated Phase 2 fuel could be subjected to heating tests. For both Phase 1 and Phase 2 irradiation tests, test conditions were tightly controlled; hence, the temperature, burnup, and fast fluence conditions experienced by the fuel samples are well known.

The conduct and results of these irradiations are discussed in the following paragraphs, grouped as AVR irradiations and MTR irradiations (Phase 1 and 2). The primary sources of data for this section as well as the data on heating test experience in paragraph 3.3.3 are Forschungszentrum Jülich reports [13], [14], [15], IAEA-TECDOC-978 [7], and archived spreadsheet data [16]. Normal operation fuel performance data for the core average fuel service conditions are used to determine the failed particle fraction assumed in the analysis of the PBMR (PBMR models consider the coated particle failure fraction and the spatial burnup, fast fluence, and temperature distribution in the core to calculate gaseous and metallic fission product release). In analyzing the German fuel experience, two different classes of particle failures are considered:

- **Exposed Kernels** – This type of failure is defined as failure of all coatings such that gases released from the kernel are also released from the particle. These failure types are a subset of the failed particles determined by the burn-leach procedure free-uranium fraction. They can be uniquely measured in unirradiated fuel by ‘weak irradiation’ in a reactor, and can be identified in irradiated spheres by Release-to-birth (R/B) ratio measurements in MTRs, and by analysis of krypton release during initial heating tests, as discussed in paragraph 3.3.3.2.
- **Silicon Carbide Defects** – This type of failure is defined by a flaw in the silicon carbide layer, either a through-wall crack or a high permeability section of the layer produced by a coating anomaly, with one or more intact dense pyrocarbon layers. These particles are also identified by the burn-leach procedure free uranium fraction, as the burn step in the procedure removes the dense pyrocarbon layers. These particles do not release gaseous fission products in normal operation or accident conditions up to 1 700 °C, but do release metallic fission products, including cesium, to a limited extent at the upper end of the operating temperature range, and to a greater degree under accident conditions.

The analysis of German fuel heating data, as discussed in paragraph 3.3.3, indicates that particles with silicon carbide defects are more prevalent than particles with exposed kernels.

3.3.2.1 AVR irradiation data

As reported in the preceding references, large number of LEU UO_2 TRISO fuel spheres were irradiated in the AVR reactor, including 24 615 GLE 3 spheres and 28 990 GLE 4 spheres (refer to Table 3 and Table 4 for characteristics). Of these, heating test data from 14 GLE 3 spheres are available and can be used to infer post-irradiation particle failure fractions. The following discussion is directed toward justification of the use of performance data from these spheres in support of the PBMR. While the burnups of the spheres irradiated in the AVR were measured and fluence can be estimated based on the measured burnup, their individual temperature history is not known. However, to reach full burnup, they were typically recycled 10 times, which would level out the variations in individual trajectories through the core, and the aggregate temperature history can be estimated with reasonable confidence (higher burnup spheres were typically recycled to the center region). During operation with a 950 °C mixed mean coolant outlet temperature, an experiment was conducted to measure the maximum temperature experienced by the fuel [11]. In the experiment, 144 spheres containing a series of encapsulated fusible wires covering a range of melting points from 920 °C to 1 280 °C were passed through the AVR core. Ninety-five spheres were loaded into the core via the central core tube, and 49 were loaded through the four peripheral tubes. The distribution of maximum fuel temperatures measured in the experiment for the central core, outer core, and combined results is shown in Figure 12. The maximum temperature experienced by a sphere is not known, as the maximum melt wire temperature was 1 280 °C; but given the distribution, it is reasonable to conclude that it was no more than 1 400 °C³. This value was used to determine the average temperature and fuel element fraction for the highest temperature points in Figure 12.

³ There is no way to determine what the peak fuel temperature actually was (a detailed 3D T&H model of the AVR core with the fuel loading it had at the time of the melt wire tests would be extremely expensive and would still retain a large degree of uncertainty). The value of 1 400 °C is based on a data consistency and plausibility argument. The main point with regard to this paper is that the temperature environment for fuel irradiated in the AVR was more severe than the fuel will experience in the PBMR. Whether the value is 1 350 °C or 1 450 °C will not alter that conclusion.

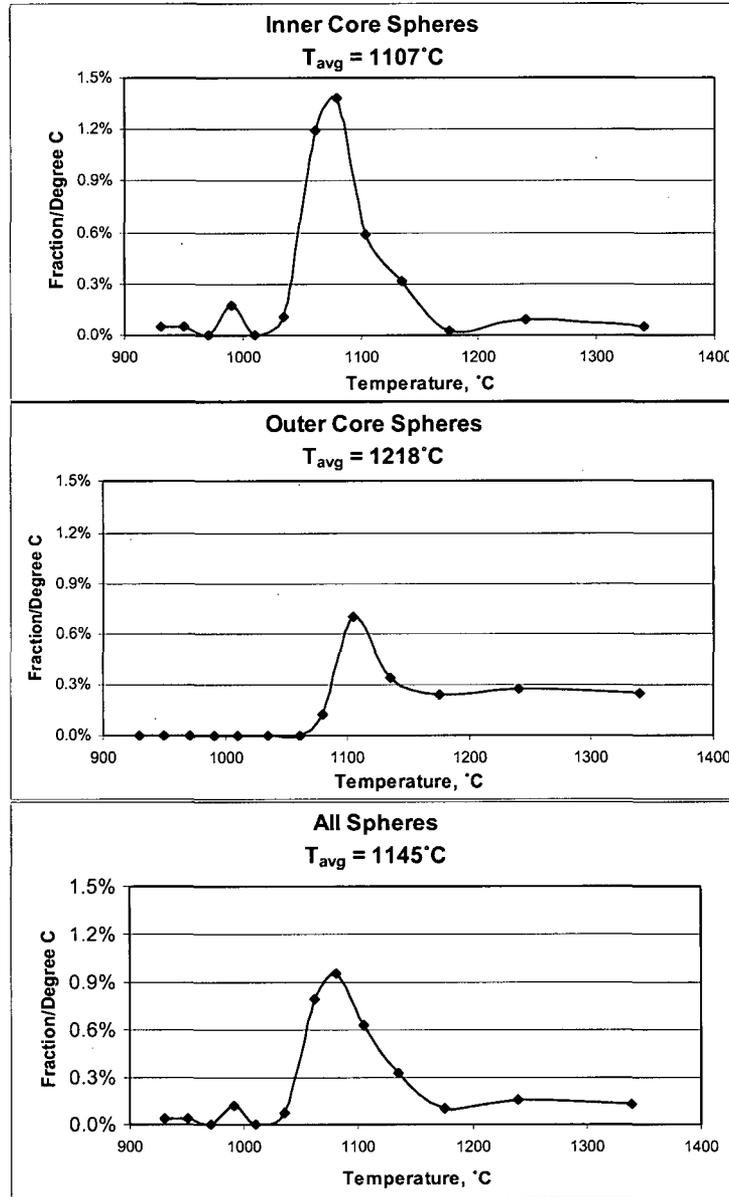


Figure 12: AVR Maximum Fuel Temperature Experiment Results

The high fuel temperatures observed in the AVR were produced by a combination of the high mixed mean coolant outlet temperature (950 °C) and the large mixed mean coolant temperature rise across the reactor (675 °C). Inside the reactor vessel, a significant fraction of the coolant bypasses the core and flows through gaps between reflector blocks, component cooling channels, etc. The high coolant temperature rise of the AVR amplifies the effect of core flow bypass as well as radial and azimuthal variations in coolant flow and power density. The AVR parameters are listed in Table 5, and used to determine an ‘enthalpy rise hot channel factor’ (the ratio of maximum fuel temperature rise to reactor mixed mean coolant ΔT) for the AVR, and the PBMR maximum fuel temperature is estimated based on assuming the same factor.

Table 5: Effect of Difference in Mixed Mean Coolant Temperature Rise

Parameter	AVR	PBMR
T _{in} (mixed mean coolant), °C	275	500
T _{out} (mixed mean coolant), °C	950	900
ΔT _{reactor} , °C	675	400
T _{max fuel} , °C	~1 400	~1 167*
ΔT _{max} , °C	1 125	667*
'Enthalpy Rise Hot Channel Factor'	1.59	1.59

Note: * Estimated using same 'Enthalpy Rise Hot Channel Factor' as AVR.

As a larger reactor with a more regular geometry (PBMR annular versus AVR cylindrical with four large control rod buttresses protruding into the core), PBMR can be expected to have a lower core bypass and less variation in the integrated axial flow and power distributions. Thus the PBMR 'enthalpy rise hot channel factor' would be expected to be lower than for the AVR, resulting in a lower maximum fuel temperature than given in Table 5 (1 137 °C), consistent with the results of PBMR core design in Table 2 (1 106 °C). On the whole, the maximum temperatures and the temperature change upon passage through the core are considerably higher and more challenging for the fuel irradiated in the AVR than will be the case for PBMR; thus AVR fuel performance data can be used to conservatively project expected PBMR fuel performance.

In order to use the fuel performance data for spheres irradiated in the AVR, it is necessary to know the failure fraction at the end of the irradiation. The data from the following procedures can be used to identify failed particles in spheres irradiated in the AVR:

- **⁸⁵Kr and ¹³⁷Cs Release on Heating** – For spheres that were subjected to heating tests in the KÜFA facility following irradiation, the existence of particles with exposed kernels can be determined by the ⁸⁵Kr release upon initial heating. Particles with silicon carbide defects can be determined by the ¹³⁷Cs release upon initial heating. The heating test data are analyzed in detail in paragraph 3.3.3 and found to provide a sound basis for differentiating and quantifying exposed kernels and silicon carbide defects in fuel irradiated in the AVR. The results of analysis of ⁸⁵Kr and ¹³⁷Cs release data for GLE 3 spheres irradiated in the AVR are given in Table 13 of paragraph 3.3.3.3.4. No exposed kernels and a total of eight silicon carbide defects were identified from these 13 spheres.
- **Fission Product Profile in Fuel-Free Zone** – An additional 13 GLE 3 and six GLE 4 spheres were subjected to a PIE procedure whereby the matrix material in the fuel-free zone was progressively removed and measured to determine the fission product profile. As shown in Figure 35, if a sphere had a particle releasing cesium, the ¹³⁷Cs profile in the fuel-free layer was inverted relative to spheres without failed particles. The results of this procedure are given in Table 17. One coated particle silicon carbide defect was identified in the 13 GLE 3 spheres and no defects were identified in the six GLE 4 spheres.
- **Progressive Deconsolidation of Fuel Spheres** – Five GLE 4 spheres were subjected to progressive deconsolidation, including determining the fission product concentration in the leachate as the deconsolidation progressed. The presence of ¹³⁷Cs in the leachate was used to indicate the presence of a failed particle. As shown in Table 18, one particle with an exposed kernel was observed in the five GLE 4 spheres examined using this procedure.

The results of the last two procedures identified above support and confirm the results and conclusions of the analysis of post-irradiation heating data. However, the fission product

profile data are not calibrated and would not distinguish between an exposed kernel and a silicon carbide defect, since it is based on cesium release. Likewise, the deconsolidation leachate analysis would not identify silicon carbide defects with an intact dense pyrocarbon layer, and particles could potentially be damaged by the deconsolidation process. In addition, the GLE 4 spheres are less prototypical of PBMR spheres, having a significantly smaller particle packing fraction and higher enrichment. Thus these data are considered confirmatory of the heating test analysis, but are not used to quantify performance of fuel irradiated in the AVR. The AVR results of Table 13 will be used in conjunction with MTR data discussed in paragraph 3.3.2.2 to quantify normal operation performance projections based on the German experience.

3.3.2.2 Materials test reactor irradiation data

A detailed listing of German fuel specimens irradiated in MTRs is provided in Table 19 and Table 20. Irradiations were conducted in three reactors (experiment designations are shown in parentheses):

- HFR in Petten, the Netherlands (HFR-xxx);
- Siloe in Grenoble, France (SL-xxx); and
- DIDO in Jülich, Germany (FRJ2-xxx).

Fuel containing LEU UO₂ TRISO particles was irradiated in the above reactors in three fuel forms:

- full-size spheres (HFR-K3, HFR-K5, HFR-K6, FRJ2-K13, FRJ2-K15);
- small (2 cm diameter) spheres embedded in a cylinder of matrix material (HFR-P4, SL-P1); and
- cylindrical compacts (FRJ2-P27).

The irradiation locations in HFR and Siloe had harder neutron spectra relative to PBMR, resulting in higher fast fluence for a given burnup. The locations in DIDO had a softer spectrum than PBMR, resulting in low fast fluence for a given burnup. There are two primary effects of fast fluence that affect the potential for particle failure:

- Exposure to fast fluence in the inner and outer dense pyrocarbon layers produces shrinkage up to approximately 4×10^{21} n/cm², imposing compressive stresses on the silicon carbide layer, which is more rigid and dimensionally stable.
- Exposure to fast fluence in the silicon carbide layer reduces its tensile strength

In irradiations conducted in the United States in the 1990s, the pyrocarbon properties were such that large failure fractions were observed as a result of excessive pyrocarbon shrinkage leading to cracking of both the pyrocarbon and silicon carbide layers [17]. As discussed in [17], the properties of the German pyrocarbon layers are such that this type of failure is not observed. However, the German dense pyrocarbon layers are observed to shrink, imposing a compressive stress on the silicon carbide layer. This shrinkage can have a beneficial effect on the structural performance of the particle by enhancing the sharing of internal pressure loads among the layers. Without pyrocarbon shrinkage, the more rigid silicon carbide layer would be subjected to most of the tensile load as fission gas pressure builds up within the particle. Thus, depending on the specific conditions, fast fluence can either increase or decrease the likelihood of particle failure. Having MTR data at both higher and lower fast fluence than PBMR provides assurance that the range of exposure conditions that can be present in PBMR fuel is enveloped by the irradiation data.

All of the irradiation data produced by the fuels listed in Table 19 and Table 20 provide useful data in assessing the performance capability of the German LEU UO₂ TRISO fuel. However, only full-sized fuel spheres of comparable enrichment and packing fraction are considered sufficiently prototypical to include in a quantified performance projection. This includes the GLE 3 spheres irradiated in the AVR and discussed previously, and the LEU Phase I and Proof Test spheres irradiated in MTRs. For the spheres subjected to heating tests, the silicon carbide defects could be identified based on the initial ¹³⁷Cs release as discussed in paragraph 3.3.3.3 and listed in Table 13. For the spheres not subjected to heating tests, no data are available on silicon carbide defects. However, the existence of particles with exposed kernels would be registered by the in-pile gaseous fission product release for fuels irradiated in MTRs. The data in Table 3-3 of [9] show that the maximum end of irradiation ^{85m}Kr R/B ratio for these spheres was 3×10^{-7} , indicating that none of the spheres listed in Table 6 contained an exposed kernel.

Table 6: Prototypical Sphere Materials Test Reactor Irradiations

ID	No. of Particles	Burnup	Fast Fluence	Irradiation Temperature	Test Temperature	Exposed Kernels	SiC Defects
MTR Spheres Subjected to Heating Tests							
HFR-K3/1	16 400	7.5	4	1 200	1 600	0	0
FRJ2-K13/2	16 400	8	0.2	1 150	1 600	0	1
FRJ2-K13/4	16 400	7.6	0.2	1 120	1 600	0	0
HFR-K3/3	16 400	10.6	5.9	920	1 800	0	0
Subtotal Particles	65 600	8.4	2.6	1098	< Avg. Subtotal >	0	1
MTR Spheres Not Subjected to Heating Tests							
HFR-K3/2	16 400	10	5.8	920		0	N/D*
HFR-K3/4	16 400	9	4.9	1 220		0	N/D
HFR-K5/1	14 580	7.8	4	923		0	N/D
HFR-K5/2	14 580	10.1	5.8	909		0	N/D
HFR-K5/3	14 580	10.3	5.9	903		0	N/D
HFR-K5/4	14 580	9.3	4.9	921		1	N/D
HFR-K6/1	14 580	8.3	3.2	1 090		0	N/D
HFR-K6/2	14 580	10.6	4.6	1 130		0	N/D
HFR-K6/3	14 580	10.9	4.8	1 140		0	N/D
HFR-K6/4	14 580	9.9	4.5	1 130		2	N/D
FRJ2-K13/1	16 400	7.5	0.2	1 125		0	N/D
FRJ2-K13/3	16 400	7.9	0.2	1 150		0	N/D
Exposed Kernels							
Total Particles	247 800	8.5	3.8	1 062	← Avg. Total →	3	n/a**
Silicon Carbide Defects							
Total Particles	65 600	8.4	2.6	1 098	← Avg. Total →	n/a	2

Notes:

* N/D – Not Determined (in all cases the cesium release into the surrounding graphite was low enough to demonstrate zero or near-zero defects).

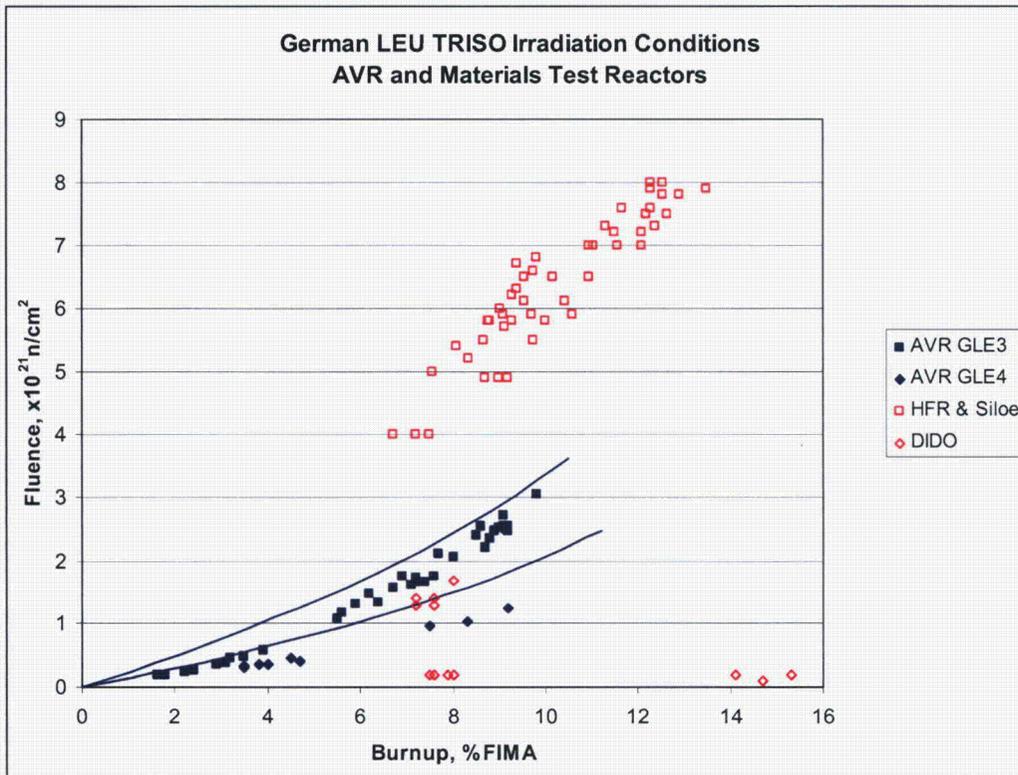
** n/a – not applicable.

3.3.2.3 Summary of normal operation irradiation experience

PBMR fuel performance analysis methods for normal operation require particle failure fractions for the average burnup, fluence, and temperature history of the fuel in the core as an input parameter. Experience with German LEU UO₂ TRISO fuel applicable to PBMR normal operation includes time varying conditions associated with actual reactor operation and controlled conditions in MTRs. Actual reactor operation in the AVR enveloped PBMR conditions in terms of mixed mean coolant outlet temperature and temperature rise across the reactor, and exposed the fuel spheres to cyclical power and temperature variations. While the actual temperature history of individual spheres irradiated in AVR is not known, the variations in temperature history would have been substantially reduced by the multiple cycles through the reactor experienced by most of the AVR spheres. As an aggregate data set, the spheres irradiated in AVR would have experienced substantially more challenging temperature histories than are expected for PBMR fuel, as discussed in paragraph 3.3.2.1.

The MTR data include known temperature histories and extremes in burnup, fluence, and time at temperatures that are well beyond the service conditions of PBMR fuel. These data also provide insights that support interpretation of the AVR irradiation data (e.g., particles with exposed kernels present from the beginning of the irradiations).

All of the German fuel AVR and MTR irradiation data discussed in paragraphs 3.3.2.1 and 3.3.2.2 are shown in Figure 13. The AVR fast fluence values were determined by a correlation with burnup, and individually adjusted to reflect the expected ±10% variation due to different trajectories taken by individual spheres. As shown in Figure 13, the aggregate envelope of the existing data on German LEU UO₂ TRISO fuel substantially exceeds the PBMR operating envelope in terms of burnup and fast fluence.



The average and maximum values of the AVR and MTR prototype sphere data as well as the combined data sets are compared with the PBMR service conditions in Table 7. It is clear from Table 7 that the prototype sphere German LEU UO₂ TRISO irradiation data in aggregate substantially exceed PBMR service conditions with the exception of maximum burnup, which is addressed in tests not included in these data, and will be addressed in the PBMR irradiation and testing program discussed in paragraph 3.4.

Table 7: Comparison of Fuel Irradiation Data Service Conditions with PBMR

Parameter	AVR Data	MTR Data	Combined	PBMR
Average Burnup, % FIMA	8.2	8.5	8.4	5.5
Maximum Burnup, % FIMA	9.8	10.6	10.6	11.2
Average Fast Fluence, x 10 ²¹ n/cm ²	2.2	3.8	3.1	1.4
Maximum Fast Fluence, x 10 ²¹ n/cm ²	2.9	5.9	5.9	3.6
Average Center Temperature, °C	N/D	1062	N/D	904
Maximum Center Temperature, °C	~1 350	1 220	~1 350	1 106
Number of Particles	229 600	247 800	477 400	

The overall German LEU UO₂ TRISO prototype sphere data set for normal operation is given in Table 8. These data are for exposed kernels and silicon carbide defects of irradiated fuel, thus they would include both manufacturing defects and in-service failures. The basis for determining the number of particles with exposed kernels and with silicon carbide defects for the spheres irradiated in the AVR and in MTRs that underwent heating tests is discussed in detail in paragraph 3.3.3. The identification of exposed kernels for spheres in Table 8 that did not undergo heating tests was made by analysis of in-pile gaseous fission product release [15]. As can be seen from Table 8, the 50% confidence silicon carbide defect fraction of 3.5×10^{-5} , which would include exposed kernels, is very close to the mean as-manufactured free uranium fraction for the combined fuels of approximately 4.7×10^{-5} (refer to Table 4), and the as-manufactured value is well within the 95% confidence value for silicon carbide defect fractions in the irradiated fuel of 5.6×10^{-5} . This provides confidence in the identification of silicon carbide defects from the initial cesium release response in the heating tests. Also, it is clear that the fraction of particles whose radionuclide retention capability is degraded in normal operation is exceedingly low. This should not be surprising given that during normal operation, the fuel must retain the capability to withstand accident conditions with very limited particle failure. Data regarding performance under accident conditions are addressed in paragraph 3.3.3.

Table 8: Normal Operation Prototype Sphere Irradiation Data Summary

ID	No. of Particles	Burnup	Fast Fluence	Irradiation Temperature	Test Temperature	Exposed Kernels	SiC Defects
AVR Spheres*							
AVR 88/15	16 400	8.7	2.4		1 600	0	0
AVR 82/20	16 400	8.6	2.4		1 600	0	1
AVR 88/33	16 400	8.5	2.3		1 600	0	3
AVR 71/22	16 400	3.5	0.5		1 600	0	0
AVR 82/9	16 400	8.9	2.5		1 600	0	N/D*
AVR 90/20	16 400	9.8	2.9		1 620T	0	0
AVR 90/2	16 400	9.2	2.7		1 620T	0	1
AVR 90/5	16 400	9.2	2.7		1 620T	0	0
AVR 85/18	16 400	9.15	2.6		1 620T	0	0
AVR 89/13	16 400	9.1	2.6		1 620T	0	0
AVR 74/11	16 400	6.2	1.4		1 700	0	1
AVR 91/31	16 400	9.0	2.6		1 700T	0	0
AVR 88/41	16 400	7.6	1.9		1 800	0	2
AVR 76/18	16 400	7.1	1.7		1 800	0	0
Materials Test Reactor Spheres							
HFR-K3/1	16 400	7.5	4	1 200	1 600	0	0
FRJ2-K13/2	16 400	8	0.2	1 150	1 600	0	1
FRJ2-K13/4	16 400	7.6	0.2	1 120	1 600	0	0
HFR-K3/3	16 400	10.6	5.9	920	1 800	0	0
HFR-K3/2	16 400	10	5.8	920		0	N/D
HFR-K3/4	16 400	9	4.9	1220		0	N/D
HFR-K5/1	14 580	7.8	4	923		0	N/D
HFR-K5/2	14 580	10.1	5.8	909		0	N/D
HFR-K5/3	14 580	10.3	5.9	903		0	N/D
HFR-K5/4	14 580	9.3	4.9	921		1	N/D
HFR-K6/1	14 580	8.3	3.2	1 090		0	N/D
HFR-K6/2	14 580	10.6	4.6	1 130		0	N/D
HFR-K6/3	14 580	10.9	4.8	1 140		0	N/D
HFR-K6/4	14 580	9.9	4.5	1 130		2	N/D
FRJ2-K13/1	16 400	7.5	0.2	1 125		0	N/D
FRJ2-K13/3	16 400	7.9	0.2	1 150		0	N/D
Analysis Summary of Irradiated Spheres							
Parameter	Number of Particles	Total Particles	Maximum Parent Population Particle Fraction				
Confidence Level that Indicated Particle Fraction is not Exceeded in Parent Population			50%	95%	97.5%		
Exposed Kernels	3	477 400	7.7×10^{-6}	1.6×10^{-5}	1.8×10^{-5}		
Silicon Carbide Defects	9	278 800	3.5×10^{-5}	5.6×10^{-5}	6.1×10^{-5}		

Note: * N/D – Not Determined.

3.3.3 Heating Test Experience

Data provided in this paragraph are taken from [13], [14], [15], [7], and [16]. Irradiated fuel spheres and compacts from the population described in paragraph 3.3.2 were subjected to heatup tests to simulate fuel performance during heatup accidents. The heating tests covered a range of temperatures up to 2 500 °C, far above temperatures achievable under accident conditions in the PBMR. As discussed in more detail below, heating tests demonstrated that at the maximum PBMR accident temperatures in the range from 1 600 °C to 1 700 °C for a few tens of hours, only a very small number of coated particles failed and ^{137}Cs and ^{90}Sr release fractions remained low.

Isothermal heating tests performed at temperatures in the range from 1 800 °C to 2 100 °C, and temperature ramp tests to 2 500 °C showed that there were no 'cliff edges' (i.e., sudden or unexpected changes) in the failure behaviour of coated particles at high temperatures; below which no failures were observed and above which all coated particles would fail uniformly. Rather, a small fraction of a percent of particle failures was observed as temperatures remained above 1 600 °C for extended periods of time (many days). This behaviour is a result of statistical variations in particle properties, particularly kernel diameter, coating thicknesses and sphericity. Outlier particles (e.g., large kernel, thin coatings, highly aspherical) will fail earlier in time and at less demanding conditions than the main population of particles with properties near the nominal values. At more elevated temperatures, i.e., above 2 000 °C, widespread particle failures and elevated releases were observed in the test specimens over periods of several hours. This is due to the onset of SiC thermal decomposition [18].

3.3.3.1 Heating test facilities and procedures

Irradiated fuel from the AVR and MTRs was subjected to heating tests under controlled conditions with the ability to quantitatively measure the release of both gaseous and metallic fission products. The basic components of the heating test facilities are shown in Figure 14, which is based on the KÜFA facility. The initial inventories of the isotopes of interest were established by measurement or calculation before starting the test, and the quantities released during the test were trapped and measured to determine the release fraction at a large number of time steps.

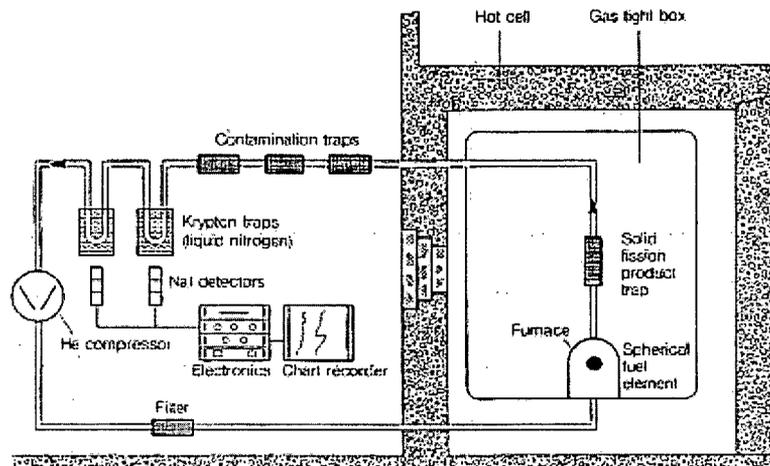


Figure 14: Heating Test Facility Schematic [14]

Two heating test facilities were used to conduct the tests. The KÜFA facility, shown in Figure 15, was used to conduct the majority of heating tests on spheres irradiated in the AVR, and all of the fuel specimens irradiated in MTRs. In addition to externally trapping and measuring the ^{85}Kr release, it had the capability to remove and replace the cooled in-furnace deposition plates during the course of the test, to allow measurement of the time-dependent release of metallic fission products. The tantalum furnace of the KÜFA facility limited temperatures to a maximum of 1 800 °C. Time-dependent temperature control allowed tests simulating the time profile of limiting DLOFC events, as well as isothermal tests at elevated temperature, to be conducted.

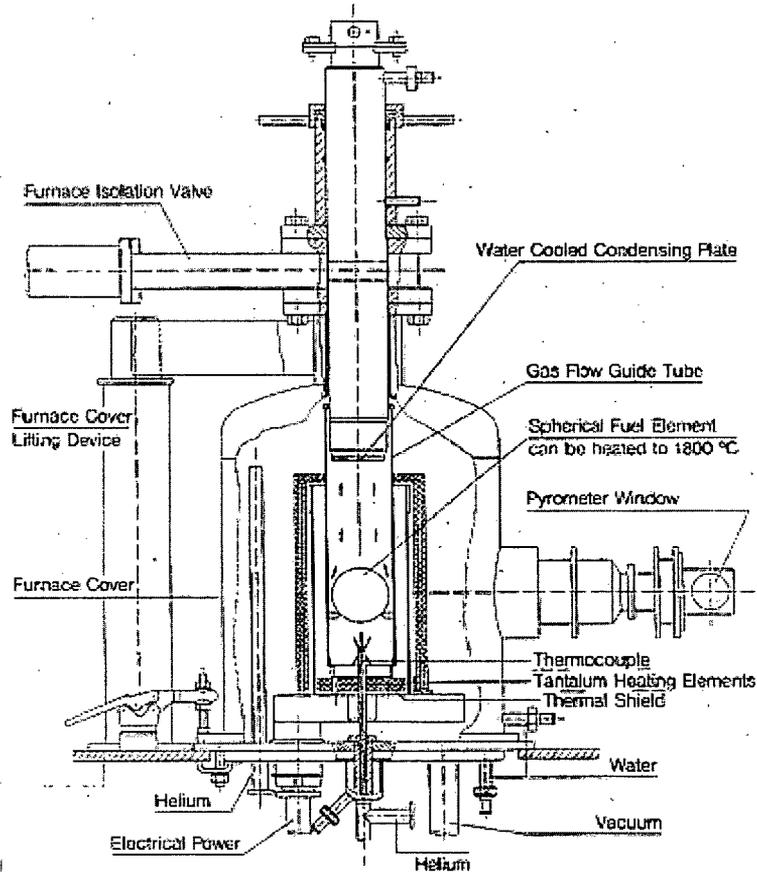


Figure 15: KÜFA Heating Test Facility [14]

An earlier test facility, labelled the 'A'-test graphite furnace, shown in Figure 16, was capable of conducting tests up to 2 500 °C. This facility was used to conduct isothermal testing on irradiated LEU UO₂ TRISO spheres at temperatures from 1 800 °C to 2 100 °C. It was also used to conduct tests with linearly increasing temperature from 1 250 °C to 2 500 °C at a nominal rate of 47 °C per hour. An important limitation of this facility was the inability to collect time-dependent metallic fission product release. In the absence of replaceable deposition plates, the fractional release of solid fission products was only available at the end of the test. In addition, the temperature, which was controlled by a pyrometer, may have been significantly higher than intended during some of the tests due to the effect of fouling of the pyrometer window.

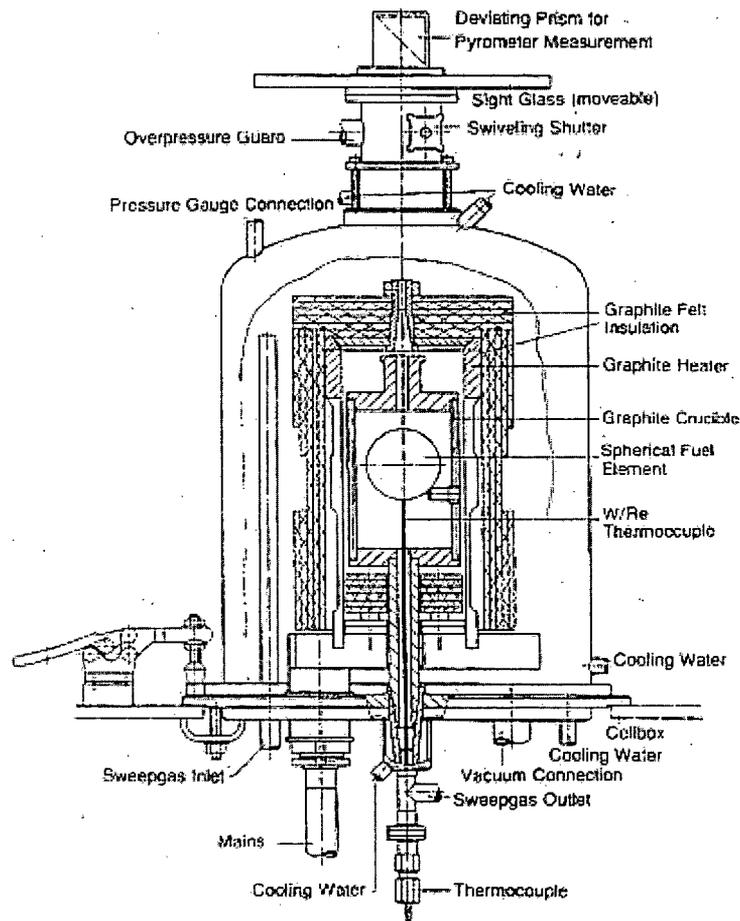


Figure 16: 'A'-Test Heating Test Facility [14]

A standard procedure was developed to establish a common initial condition prior to starting the elevated temperature testing, as depicted in Figure 17. The hold at 300 °C provided for cleanup of the helium circuit and removal of moisture from graphite components. Holds at 1 050 and 1 250 °C were for simulation of operating temperatures and equilibration of fuel and fission products. The heatup rate to the test temperature reflects the thermal characteristics of HTRs.

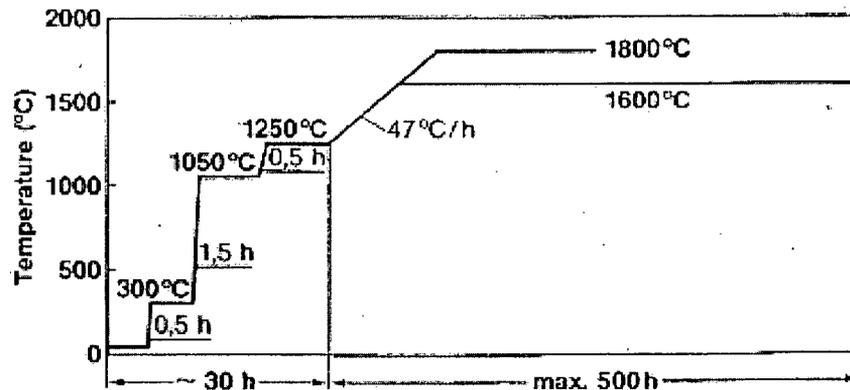


Figure 17: Temperature Profile in Standard Heating Test [14]

The results of the heating tests conducted on German LEU UO₂ TRISO fuel are summarized in the following paragraphs.

3.3.3.2 Heating test results – ⁸⁵Kr release (exposed kernel failure fraction)

A primary input to the accident analysis is the exposed kernel failure fraction as a function of temperature. At the elevated temperatures of the heating tests, a substantial fraction of the long-lived isotope ⁸⁵Kr diffuses out of the kernel and is available for rapid release if the coating layers fail. Thus the ⁸⁵Kr fractional release data typically give a clear indication of coating failure during the test. In the figures that follow, a line indicating 100% release from one particle is shown. This value is 1/(number of particles in the sphere = 16 400) or 6.1×10^{-5} , and applies to all isotopes. The ⁸⁵Kr release results as a function of temperature are discussed in the following paragraphs.

3.3.3.2.1 1 600 °C isothermal and transient testing

A large number of post-irradiation heating tests of fuel spheres with a maximum temperature of approximately 1 600 °C were conducted, mainly in support of the HTR Modul design and licensing. The test fuels included fuel irradiated in the AVR as well as in Petten (HFR) and DIDO (FRJ2), and the tests included both isothermal tests at 1 600 °C and transient simulations with a maximum temperature of 1 620 °C. Burnups ranged from 3.5% to 9.8% FIMA. The isothermal ^{85}Kr release data for the fuel spheres irradiated in both the AVR and in MTRs are given in Figure 18.

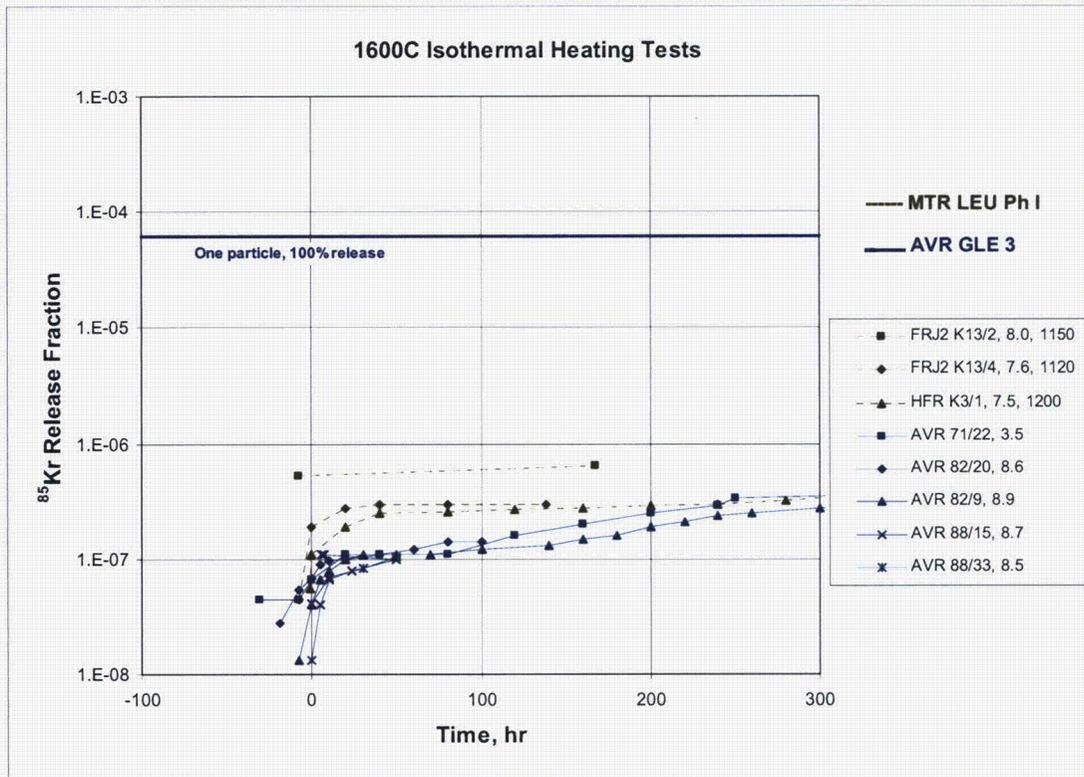


Figure 18: 1 600 °C Isothermal Testing ^{85}Kr Release

The legend in Figure 18 provides the identifier for each fuel test specimen (FRJ2 refers to irradiation in DIDO, HFR to Petten), along with the end of irradiation burnup and, for the MTR irradiations, the irradiation temperature. The tests are listed in the legend approximately in order of decreasing release at 100 h. The theoretical release fraction resulting from 100% release from a coated particle is shown for comparison. While the data are shown for up to 300 h, a duration of 100 h is considered sufficient to envelope the range of PBMR heatup accidents. No particle failures were indicated in any of the spheres tested, with the variation in levels of release associated with variation in levels of heavy metal contamination fraction among the spheres.

Additional 1 600 °C isothermal tests of irradiated fuel particles were conducted on particles from the same coating batch as was used for the LEU TRISO Phase I spheres (EUO 2308). These particles were in a different fuel form, contained in a fuelled sphere 2 cm in diameter, located within a cylinder of matrix material. The fuelled zone was formed in an isostatic press and surrounded by a 2 cm layer of matrix to form a sphere that was then machined to a

cylinder of dimensions as required by the HFR and Siloe irradiation test rigs. In the data presented here, the burnups for HFR-P4 and SL-P1 were adjusted to reflect the results of burnup measurements of selected fuel specimens at Seibersdorf [23]. This resulted in a reduction of the KFA measurements of approximately 13%. Although there is not a consensus regarding whether the Seibersdorf or the KFA measurements are more credible, it was considered conservative for the purposes of this paper to adjust to the Seibersdorf results. The heating test results, along with burnup, fluence, and irradiation temperature, are shown in Figure 19.

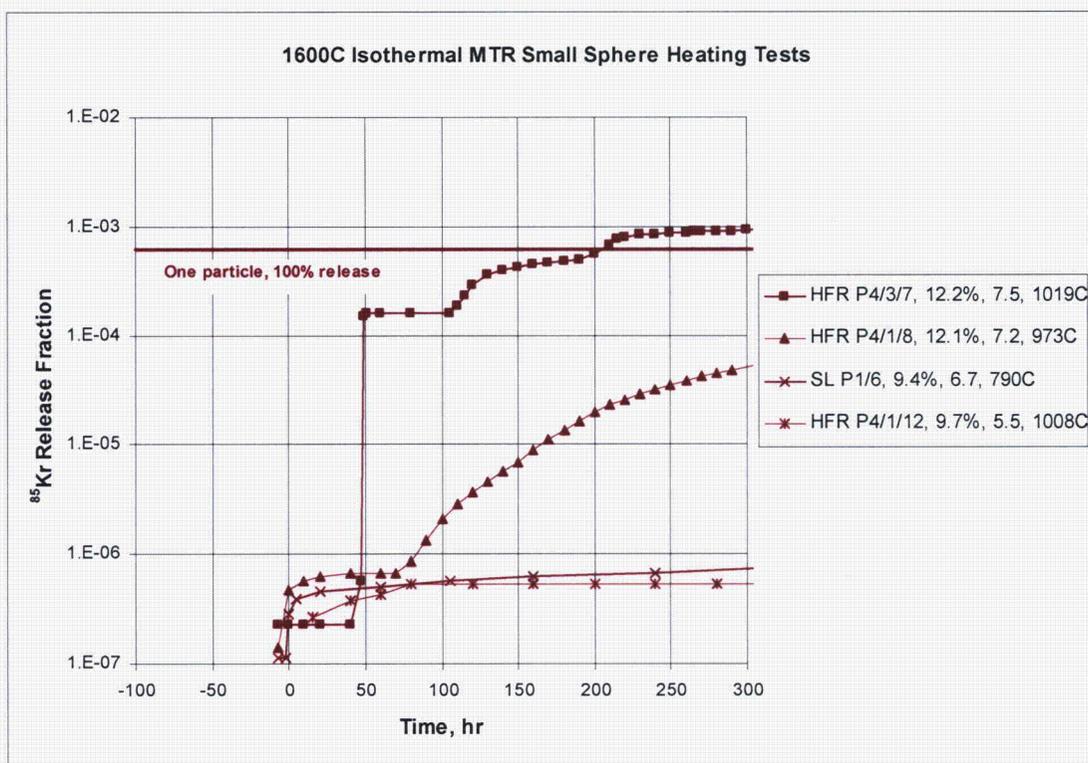


Figure 19: 1 600 °C Isothermal Testing, Small Spheres ^{85}Kr Release

The theoretical release fraction resulting from 100% release from a coated particle is shown for the small spheres from HFR-P4 and SL-P1. The small spheres have a higher release fraction for a single particle failure because they have fewer particles than full-sized spheres (1 631 in the HFR P4 spheres and 1 666 in the SL P1 spheres).

The HFR-P4 irradiation rig contained three axial stacks or 'legs' of small spheres. HFR-P4 leg 1 and leg 3 contained small spheres with particles from coating batch EUO 2308, and leg 2 contained small spheres with particles from batch EUO 2309 having 51 μm thick SiC. The results from leg 2 were not included in this assessment because of the different particle design. The results may indicate a dependence of heating test performance on the combination of burnup, fast fluence, and irradiation temperature at these elevated values.

Several fuel spheres irradiated in the AVR were subjected to a simulated thermal transient with a maximum temperature of 1 620 °C, with results as shown in Figure 20 (burnup for each sphere is indicated in the legend). The temperature traces are included, showing the close replication of the temperatures among the tests.

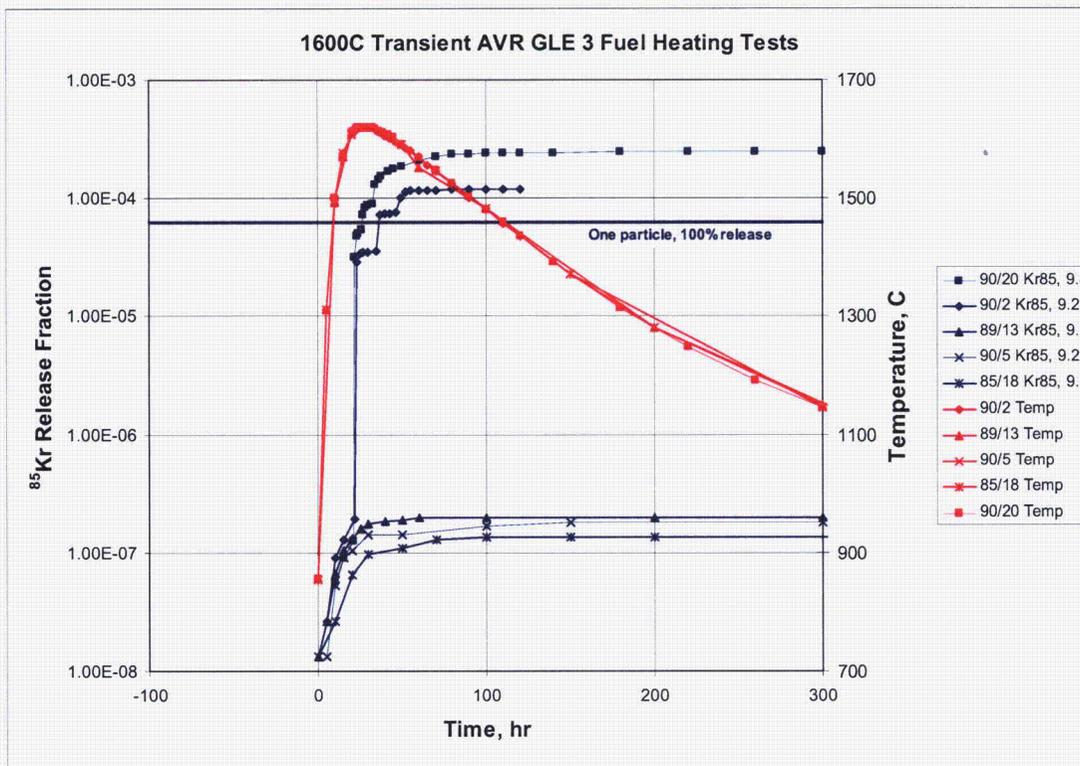


Figure 20: 1 600 °C Transient Temperature Testing ⁸⁵Kr Release

In some of these tests, there is an indication of progressive failure of particles, with sudden increases followed by longer-term releases. Assuming 100% release from the failed particles results in an estimate of four failed particles in AVR 90/20 and two failed particles in AVR 90/2. Again, the tests are listed in order of decreasing release in the legend. It is noteworthy that the releasing particles are from the spheres with the highest burnup, and that all of the AVR spheres in this test had higher burnups than those in the isothermal tests of Figure 18. The elevated temperatures likely experienced by fuel irradiated in the AVR, as discussed in paragraph 3.3.2.1, in conjunction with the higher burnups, may be a factor.

The small spheres irradiated in HFR-P4 and SL-P1 provide valuable data with regard to particle performance to high burnups. However, they represent a substantial departure from the geometry of the full-sized PBMR spheres, and thus are not included in the performance statistics regarding normal operation (no failures were observed during irradiation, thus their inclusion would reduce the failure fraction). The overall results of the 1 600 °C testing of the prototypical GLE 3 and LEU Phase I spheres are summarized in Table 9. In summary, for the 1 600 °C heating tests, a total of six failed particles was observed out of a total of 213 200 particles. AVR spheres 88/15 and 88/33 were included with the other spheres heated for 100 h, although their test durations at 1 600 °C were 50 h. They were subsequently tested at 1 800 °C, and their inclusion is based on their performance at both temperatures.

The average burnup and fast fluence of the population of spheres subjected to heating tests at 1 600 °C (8.3 and 2.2) are considerably higher than the average PBMR burnup and fast fluence (5.5 and 1.4 as given in Table 2). Thus the data can be conservatively applied to project the performance of PBMR fuel. A possible trend between particle failure during heating testing and higher burnup and temperature during irradiation is observed, but the statistics are limited. The release profiles of the failures indicate progressive failure of

individual particles in some cases, and multiple failure modes. The maximum through-coating failure fractions from these 1 600 °C heating tests at 50%, 95%, and 97.5% confidence are 3.1×10^{-5} , 5.6×10^{-5} , and 6.1×10^{-5} respectively. It is worth noting that these failure fractions would include any failed particles resulting from manufacturing and irradiation, and the 50% confidence value is lower than the mean as-manufactured free uranium fraction for both the GLE 3 and LEU Phase 1 fuels. As noted earlier in the normal operation section (paragraph 3.3.2.3), this is not surprising, since the burn-leach process used to determine free uranium fraction identifies both particles with through-coating failures and particles with defective silicon carbide layers but intact pyrocarbon layers. The latter particles would not be identified by gaseous fission product release, but will show up in solid fission product release, as discussed in paragraph 3.3.3.3.

Table 9: Summary of 1 600 °C Heating Test Krypton Release Results

Identifier	Burnup (% FIMA)	Fast Fluence (10^{21} n/cm ²)	Irradiation Temperature (°C)	Test Type	Number of Particles	Exposed Kernels
AVR Spheres (GLE 3)						
90/20	9.8	2.94	Not available	Transient	16 400	4
90/2	9.2	2.66	Not available	Transient	16 400	2
90/5	9.2	2.66	Not available	Transient	16 400	0
85/18	9.15	2.63	Not available	Transient	16 400	0
89/13	9.1	2.61	Not available	Transient	16 400	0
82/9	8.9	2.52	Not available	Isothermal	16 400	0
88/15	8.7	2.43	Not available	Isothermal	16 400	0
82/20	8.6	2.38	Not available	Isothermal	16 400	0
88/33	8.5	2.33	Not available	Isothermal	16 400	0
71/22	3.5	0.48	Not available	Isothermal	16 400	0
MTR Spheres (LEU Phase I)						
HFR-K3/1	7.5	4	1 200	Isothermal	16 400	0
FRJ2-K13/2	8	0.2	1 150	Isothermal	16 400	0
FRJ2-K13/4	7.6	0.2	1 120	Isothermal	16 400	0
Average	8.3	2.2	Total		213 200	6
Maximum Parent Population Exposed Kernel Fraction, 50% confidence						3.1×10^{-5}
Maximum Parent Population Exposed Kernel Fraction, 95% confidence						5.6×10^{-5}
Maximum Parent Population Exposed Kernel Fraction, 97.5% confidence						6.1×10^{-5}

3.3.3.2.2 1 700 °C isothermal and transient testing

The isothermal and transient heating test of irradiated German LEU TRISO fuel to 1 700 °C was limited to two spheres irradiated in AVR and two small spheres irradiated in Siloe. One of the spheres from the AVR was subjected to a transient temperature profile with a maximum of 1 700 °C, while the remainder were subjected to an isothermal test at 1 700 °C, with results as shown in Figure 21. In the isothermal tests, it appears that the AVR sphere experienced a partial failure of one particle, while the small spheres irradiated in Siloe remained approximately three orders of magnitude below the level of a single particle failure during the first 100 h.

The AVR sphere subjected to the transient simulation experienced an estimated total of 19 particle failures, with the first failure indicated as the temperature reached 1 600 °C. The statistics for these tests are extremely limited, so it is difficult to reach any conclusions from the results. It is noteworthy that the sphere experiencing the most failure was irradiated in AVR to a significantly higher burnup than the AVR sphere in the isothermal test. While the small spheres in Siloe experienced a higher burnup, the irradiation temperatures were relatively low. Thus the relative behaviour of the tested fuel is consistent with the service conditions experienced during irradiation.

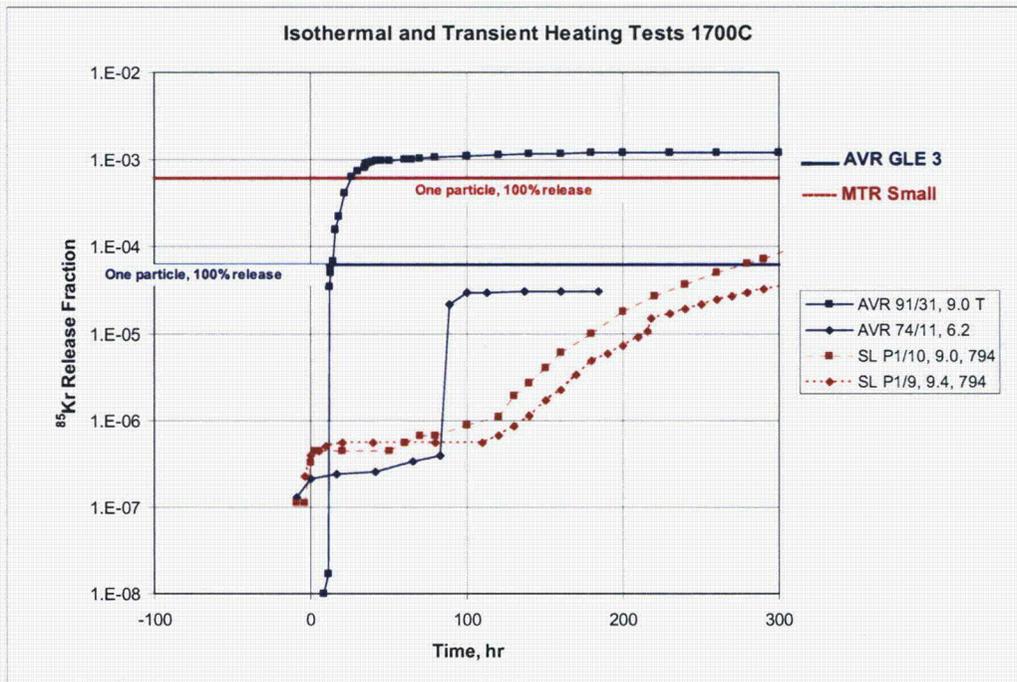


Figure 21: 1 700 °C Combined Testing ⁸⁵Kr Release

The results of the 1 700 °C testing for the GLE 3 spheres are summarized in Table 10. The results are dominated by the failures observed in sphere AVR 91/31.

Table 10: Summary of 1 700 °C Heating Test Krypton Release Results

Identifier	Burnup (% FIMA)	Fast Fluence (10 ²¹ n/cm ²)	Irradiation Temperature (°C)	Test Type	Number of Particles	Failed Particles
AVR Spheres (GLE 3)						
AVR 91/31	9.0	2.6	Not available	Transient	16 400	19
AVR 74/11	6.2	1.4	Not available	Isothermal	16 400	1
Average	7.6	2.0	Total		32 800	20
Maximum Parent Population Exposed Kernel Fraction, 50% confidence						6.3 x 10 ⁻⁴
Maximum Parent Population Exposed Kernel Fraction, 95% confidence						8.9 x 10 ⁻⁴
Maximum Parent Population Exposed Kernel Fraction, 97.5% confidence						9.4 x 10 ⁻⁴

3.3.3.2.3 1 800 °C isothermal testing

Isothermal tests at 1 800 °C were conducted on six GLE 3 spheres irradiated in the AVR, two LEU Phase I spheres irradiated in Petten and DIDO, and one small sphere irradiated in Petten. The results are shown in Figure 22 on the same time scale as the plots of the 1 600 °C and 1 700 °C results. It is apparent by comparison with the 1 600 °C isothermal results of Figure 18 that the releases are in general considerably higher and occur much earlier in the 1 800 °C testing. Also, the burnup dependency that was apparent in the lower temperature tests appears not to be present. In fact, the two highest releasing spheres had the lowest burnup. The AVR 74/10 and AVR 70/33 spheres were heated in the 'A'-test furnace, while the remainder were heated in the KÜFA facility. The following was noted in [15]:

'In some experiments performed with the 'A'-heating furnace the measured temperatures may have been underestimated. The fuel element temperatures were measured with a hand-held pyrometer through a window in the furnace. It was found that sometimes the window became colored or blackened, thus reducing the measuring effect, which is equivalent of indicating a lower temperature than was actually present.'

Thus the curves in Figure 22 for AVR 74/10 and AVR 70/33 may represent releases at significantly higher temperature than the other curves from tests conducted in the KÜFA facility. Another factor was that these two spheres were each heated in serial tests conducted several months apart, with the results combined. The two tests were retained in the data set, since their inclusion conservatively increases the calculated failure fraction.

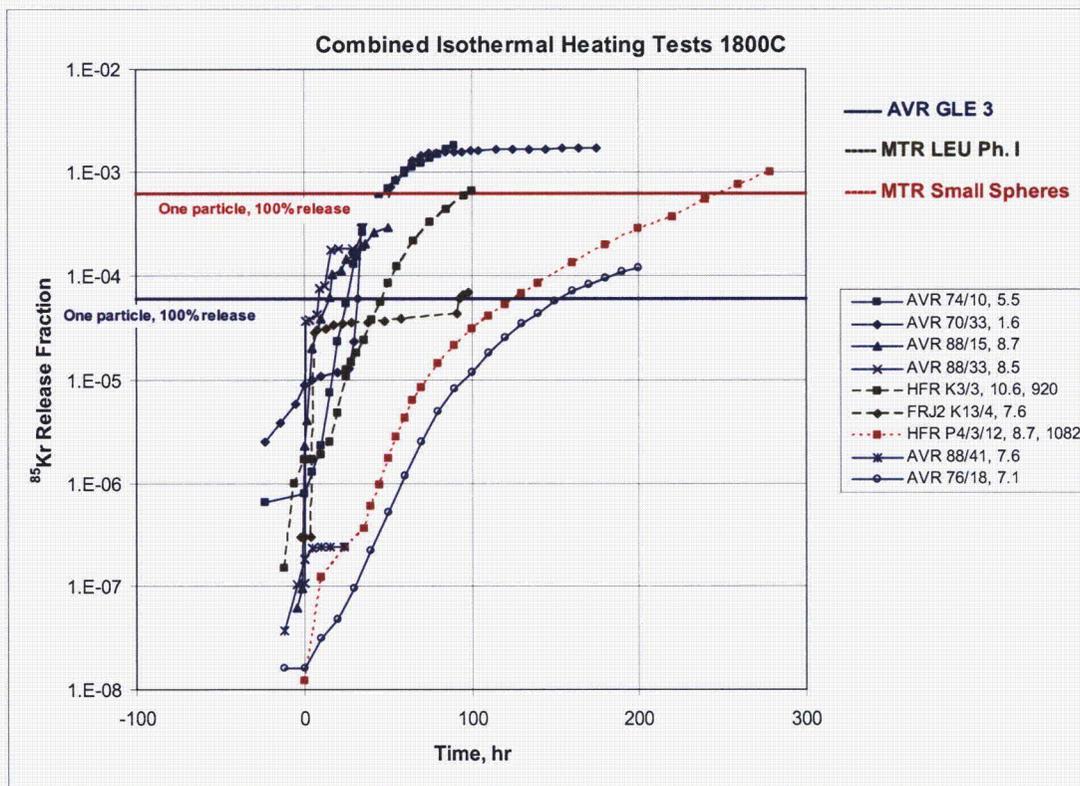


Figure 22: 1 800 °C Isothermal Testing ⁸⁵Kr Release

To better illustrate the time dependence of the 1 800 °C data, Figure 22 is repeated on a shorter time scale in Figure 23. In this figure, a sharp inflection in the AVR 70/33 release at 30 h is apparent. This is the beginning of the second 30 h heating test, which was conducted approximately three months after the first 30 h test. The sudden increase may have been the result of significantly higher temperatures in the second test due to possible temperature control errors noted previously. If the 74/10 and 70/33 data are excluded, the release trends with burnup and irradiation temperature are generally consistent with the observations at the lower temperatures.

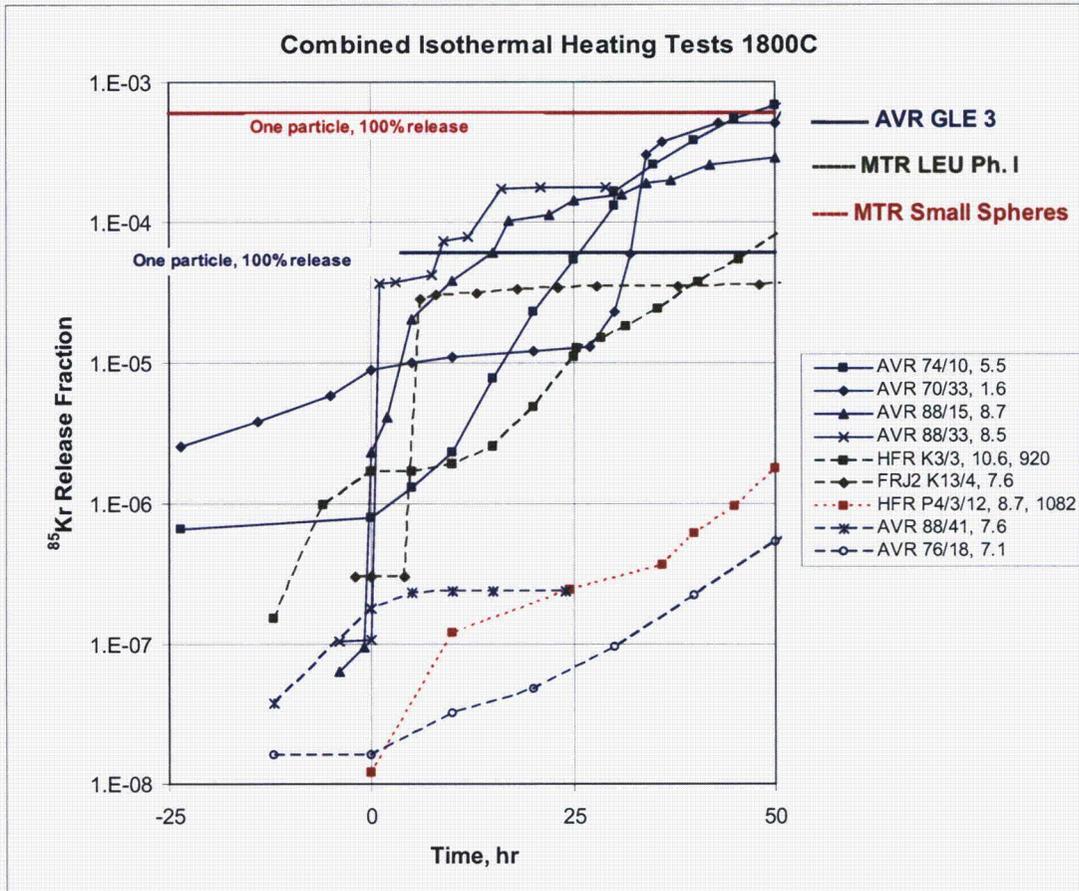


Figure 23: 1 800 °C Isothermal Testing Short Term ^{85}Kr Release

The results of the 1 800 °C heating tests for the GLE 3 and LEU Phase I spheres are summarized in Table 11 for three different heating times. The resulting 50%, 95%, and 97.5% confidence failure fractions for each of these three durations are shown. As shown in Table 11, the results are dominated by the failures in AVR 74/10 and 70/33. Since the basis for inclusion of these data is uncertain (temperatures may have been significantly higher than indicated), the results are also analyzed without them. This results in a reduction in the failure fraction at 100 h by more than a factor of two. For consistency with the results of the 1 600 °C and 1 700 °C testing, the 100 h duration data will be used for comparison in the summary discussion of the heating test results.

Table 11: Summary of 1 800 °C Heating Test Krypton Release Results

Identifier	Burnup (% FIMA)	Fast Fluence (10 ²¹ n/cm ²)	Irradiation Temperature (°C)	Test Type	Number of Particles	Failed Particles		
						30 h	50 h	100 h
AVR GLE 3								
88/15	8.7	2.4		Isothermal	16 400	3	5	NM*
88/33	8.5	2.3		Isothermal	16 400	3	NM	NM
88/41	7.6	1.9		Isothermal	16 400	0	NM	NM
76/18	7.1	1.7		Isothermal	16 400	0	0	0
74/10	5.5	1.1		Isothermal	16 400	2	11	31
70/33	1.6	0.2		Isothermal	16 400	1	8	26
MTR LEU Phase I								
HFR K3/3	10.6	5.9	920	Isothermal	16 400	0	1	11
FRJ2 K13/4	7.6	0.2	1 120	Isothermal	16 400	1	1	1
Results Including all Spheres								
Heating Time			30 h	50 h	100 h			
Average Burnup			7.2	6.9	6.5			
Average Fast Fluence			2.0	1.9	1.8			
Number Failed Particles			10	26	69			
Total Number Particles			131 200	98 400	82 000			
Exposed Kernel Fraction, 50% confidence			8.1 x 10 ⁻⁵	2.7 x 10 ⁻⁴	8.5 x 10 ⁻⁴			
Exposed Kernel Fraction, 95% confidence			1.3 x 10 ⁻⁴	3.7 x 10 ⁻⁴	1.03 x 10 ⁻³			
Exposed Kernel Fraction, 97.5% confidence			1.4 x 10 ⁻⁴	3.9 x 10 ⁻⁴	1.06 x 10 ⁻³			
Results Excluding Spheres AVR 74/10 and 70/33								
Heating Time			30 h	50 h	100 h			
Average Burnup			8.4	8.5	8.4			
Average Fast Fluence			2.4	2.6	2.6			
Number Exposed Kernels			7	7	12			
Total Number Particles			98 400	65 600	49 200			
Exposed Kernel Fraction, 50% confidence			7.8 x 10 ⁻⁵	1.2 x 10 ⁻⁴	2.6 x 10 ⁻⁴			
Exposed Kernel Fraction, 95% confidence			1.3 x 10 ⁻⁴	2.0 x 10 ⁻⁴	4.0 x 10 ⁻⁴			
Exposed Kernel Fraction, 97.5% confidence			1.5 x 10 ⁻⁴	2.2 x 10 ⁻⁴	4.3 x 10 ⁻⁴			

Note: * NM – Not Measured.

3.3.3.2.4 Higher temperature heating tests

Although the 1 800 °C tests were considered sufficient for exploration of performance margins for the HTR-Modul, additional higher temperature tests were conducted in support of other large plant designs (data from [13], [15], and [16]). These include isothermal tests at 1 900 °C, 2 000 °C, and 2 100 °C, and linearly increasing temperature up to 2 500 °C. All of these tests were conducted in the 'A'-test furnace, so the temperatures could have been higher than indicated in some cases. The results of these tests are shown in Figure 24 and Figure 25. These results show generally consistent effects of burnup and generally increasing failure fractions and release rates with temperature at temperatures far above conditions that are achievable in a PBMR. They also show that the failure and release fractions increase progressively and do not indicate any sudden shift in performance. For the 1 900 °C and 2 000 °C tests, three of the five spheres tested did not indicate particle failures. Thus even at temperatures up to 2 000 °C for 25 h, the releases are on the order of 1% for

the highest releasing spheres, and the majority of the spheres tested did not evidence particle failure.

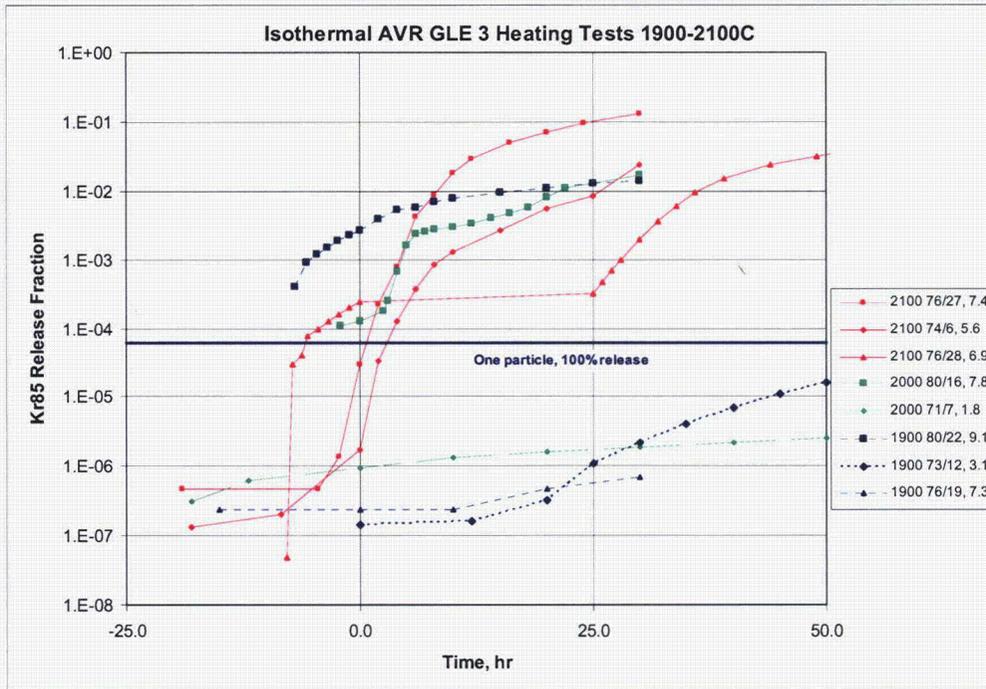


Figure 24: 1 900 to 2 100 °C Isothermal Testing ⁸⁵Kr Release Results

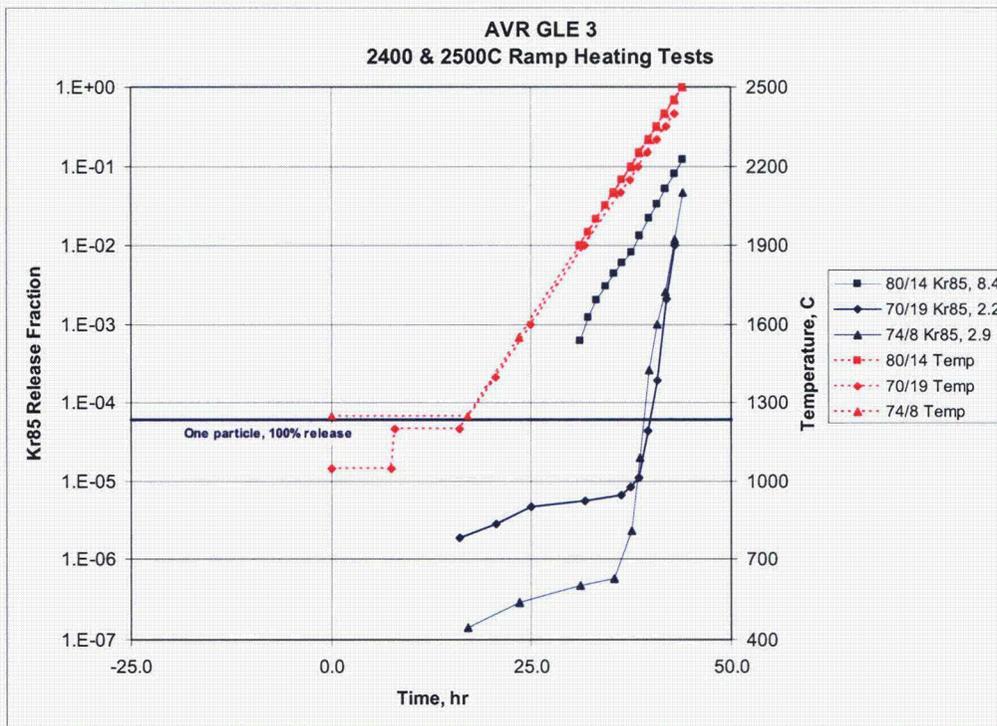


Figure 25: Temperature Ramp Testing ⁸⁵Kr Release

3.3.3.2.5 Summary of heating test krypton release results

All of the heating tests included in the statistics of the preceding paragraphs utilized full-sized spheres from either the GLE 3 or LEU Phase I fuel campaigns. As shown in Table 4, the characteristics of these fuel types are almost identical to the PBMR fuel specification. There is only a small difference in the fuel particle loadings in the spheres: 16 400 particles for the GLE 3 and LEU Phase I fuels and 14 440 for the PBMR specification. Another difference is the use of A3-27 matrix in the test fuel as compared to the A3-3 matrix that will be used for the PBMR fuel. The A3-3 matrix formulation was also used in the HTR-K5 and HTR-K5 spheres, which were irradiated but not subjected to heating tests in the German program. This difference is not expected to alter the fuel performance, which will be confirmed in the planned testing of PBMR fuel.

The quantitative results of the heating tests are summarized in Table 12. The average burnup and fast fluence of the populations of particles included in the heating tests are considerably higher than the PBMR core average burnup of 5.5% FIMA and core average fast fluence of 1.4×10^{21} n/cm², as reported in Table 12.

Table 12: Summary of Heating Test Krypton Release Results

Parameter	Test Temperature			
	1 600 °C	1 700 °C	1 800 °C	1 800 °C*
Average Burnup, % FIMA	8.3	7.6	6.5	8.4
Average Fast Fluence, 10^{21} n/cm ²	2.2	2.0	1.8	2.6
Number Particles	213 200	36 132	8 3631	50 831
Number Exposed Kernels	6	20	69	12
Exposed Kernel Fraction** (50% confidence)	3.1×10^{-5}	5.72×10^{-4}	8.33×10^{-4}	2.49×10^{-4}
Exposed Kernel Fraction** (95% confidence)	5.6×10^{-5}	8.04×10^{-4}	1.01×10^{-3}	3.82×10^{-4}
Exposed Kernel Fraction** (97.5% confidence)	6.1×10^{-5}	8.55×10^{-4}	1.04×10^{-3}	4.12×10^{-4}

Notes:

* Excludes AVR 74/10 and 70/33 test data.

** Maximum parent population exposed kernel fraction.

3.3.3.3 Heating test results – metallic fission product release

The primary metallic fission products of radiological interest for LEU UO₂ TRISO fuel under PBMR service conditions, due to a combination of their yield as well as thermochemical and nuclear properties, include silver, cesium, and strontium. Their relative characteristics as they relate to the radiological profile of the PBMR are summarized here:

- **Silver** is known to begin to be released from TRISO fuel at temperatures near the upper end of PBMR normal operating temperatures, and to be released from intact particles in significant fractions after several days at 1 600 °C. It deposits in graphite at ~900 °C and plates out on metallic surfaces at relatively high temperatures (~800 °C) and diffuses into the substrate, where it is effectively captured. As a result, silver does not present an important concern with regard to offsite dose, but can be a dominant contributor to occupational dose, depending on component maintenance requirements.
- **Strontium** is retained in oxide kernels during normal operation even when coatings are defective, and is slowly released at elevated temperatures beyond PBMR accident conditions (i.e. approaching 1 800 °C for several days). Additionally, it is absorbed in

matrix graphite and little is released from spherical fuel elements. Since strontium does not emit gammas, it is difficult to measure and the results shown have an uncertainty of an order of magnitude, while the other fission product release data have an uncertainty of around 10%.

- **Cesium** is also released from particles with defective silicon carbide layers in normal operation and can be released from the fuel spheres depending on the local conditions. However, there is a significant delay in release from the sphere due to holdup in the matrix.

Since they can be released from particles with intact pyrocarbon layers, and have significant lag times for release from a sphere if a through-coating failure develops, metallic fission products are not good indicators of through-coating failures during the heating tests. However, they provide important data on the presence of particles with defective silicon carbide layers and intact pyrocarbon layers. In order to illustrate the relative behaviours of silver, strontium, and cesium, data on releases of all three are discussed for representative heating tests at 1 600 °C, 1 700 °C, and 1 800 °C. Cesium release is also presented in conjunction with krypton release for all of the available GLE 3 and LEU Phase I sphere tests at 1 600 °C, 1 700 °C, and 1 800 °C. The primary cesium isotopes of interest are ^{134}Cs and ^{137}Cs , whose inventories developed differently as a function of burnup, but whose release fractions under accident conditions are very similar. Thus to maintain focus and limit the amount of data presented, the ^{137}Cs releases will be provided, recognizing that the ^{134}Cs data are also available. The combination of ^{85}Kr and ^{137}Cs data in the early phase of the heating tests allows determination of both through-coating failure fractions and failure fractions of particles with silicon carbide defects and an intact pyrocarbon layer. The ^{85}Kr behaviour was discussed in paragraph 3.3.3.2, but is also included here to contrast with the ^{137}Cs release.

3.3.3.3.1 1 600 °C isothermal and transient testing

The $^{110\text{m}}\text{Ag}$, ^{90}Sr and ^{137}Cs release data for a representative 1 600 °C isothermal heating test are presented in Figure 26 (data from [15]). At 1 600 °C, the silver is released relatively slowly from intact particles. With the exception of low-level initial release likely resulting from contamination near the surface, the strontium is retained within the sphere, although some release may be occurring from the particles, and the cesium is retained within the particles. (At this temperature, if cesium were being released from intact particles, a significant fraction would be released from the spheres.)

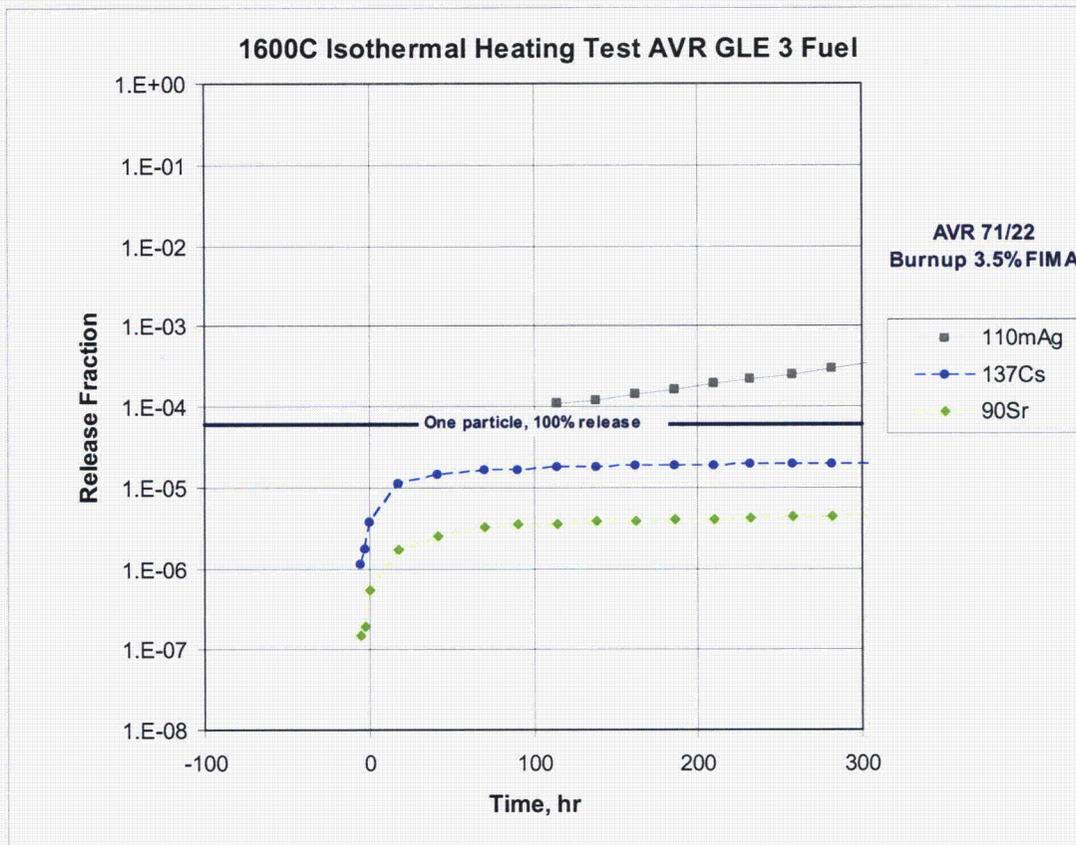


Figure 26: AVR 71/22 1 600 °C Isothermal Testing ^{110m}Ag, ⁹⁰Sr and ¹³⁷Cs Release

The ⁸⁵Kr and ¹³⁷Cs 1 600 °C isothermal heating test results for the AVR GLE 3 fuel are shown in Figure 27. As with earlier plots, the legend includes the sphere burnup in % FIMA. The primary value of these data is the identification of particles with a defective silicon carbide layer at the end of irradiation, as indicated by the ¹³⁷Cs release at the beginning of the test. As was discussed in paragraph 3.3.3.2, the ⁸⁵Kr data show that there were no particles with through-coating failure in any of the four spheres presented. The AVR 82/9 sphere data, which were presented in paragraph 3.3.3.2, are not included here, because the ¹³⁷Cs release data were reported to have been distorted by contamination.

As shown in Figure 27, the ¹³⁷Cs release for spheres AVR 71/22 and AVR 88/15 levels out well below one particle inventory, indicating that the release is from externally deposited cesium onto the sphere surface and heavy metal contamination in the sphere (the level of ¹³⁷Cs due to heavy metal contamination at the end of irradiation is much higher than that of ⁸⁵Kr, because most of the ⁸⁵Kr from contamination would have been released from the sphere during the irradiation). The ¹³⁷Cs data for AVR 82/20 were asymptotically approaching the single particle inventory through the first 70 h. The ¹³⁷Cs release for the last data point of AVR 88/33 corresponds to two particle inventories. Since it is not clear that the curve is approaching the asymptote, the response was interpreted as three particles with defective silicon carbide layers. In summary, the ¹³⁷Cs data indicate there were no particles with silicon carbide defects in spheres 71/22 and 88/15, one in 82/20, and three in 88/33.

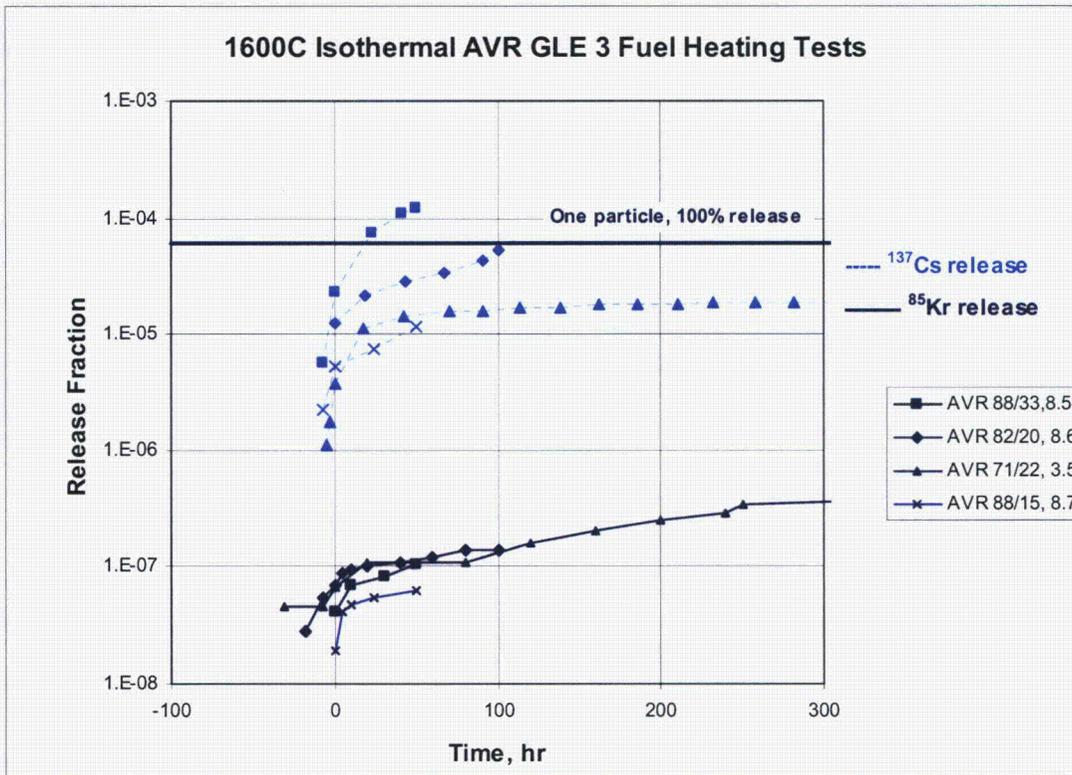


Figure 27: 1 600 °C Isothermal GLE 3 Testing 85Kr and 137Cs Release

The 85Kr and 137Cs 1 600 °C transient heating test results for the AVR GLE 3 fuel are shown in Figure 28.

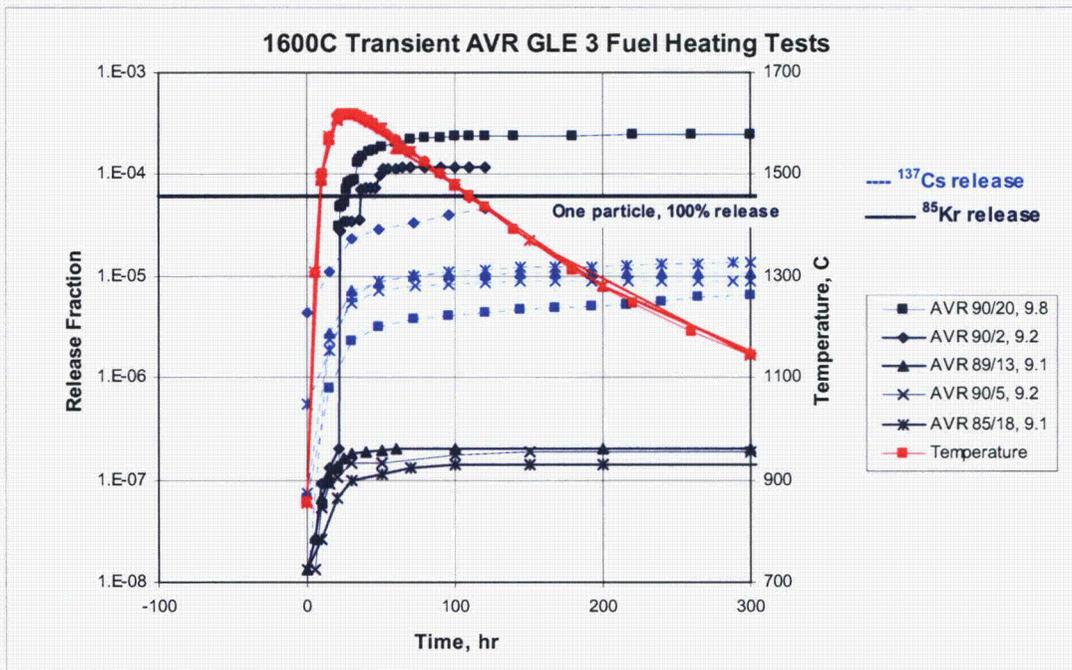


Figure 28: 1 600 °C Transient GLE 3 Testing 85Kr and 137Cs Release

The initial ^{85}Kr responses for the 1600 °C transient tests indicate there were no through-coating particles failures at the end of irradiation for any of the spheres. The ^{137}Cs responses indicate one particle with a silicon carbide defect in sphere AVR 90/2, and no particles with silicon carbide defects in the other four spheres.

The ^{85}Kr and ^{137}Cs 1600 °C isothermal heating test results for the LEU Phase I fuel are shown in Figure 29. The initial ^{85}Kr responses indicate there were no through-coating particles failures at the end of irradiation for any of the spheres. The ^{137}Cs responses indicate one particle with a silicon carbide defect in sphere FRJ2 K13/2, and no particles with silicon carbide defects in the other two spheres at the end of irradiation. The beginning of an upward trend in the cesium release from HFR K3/1 after 200 h may indicate the onset of increasing permeability in a silicon carbide layer, with the level of release remaining an order of magnitude below a single particle inventory after 300 h.

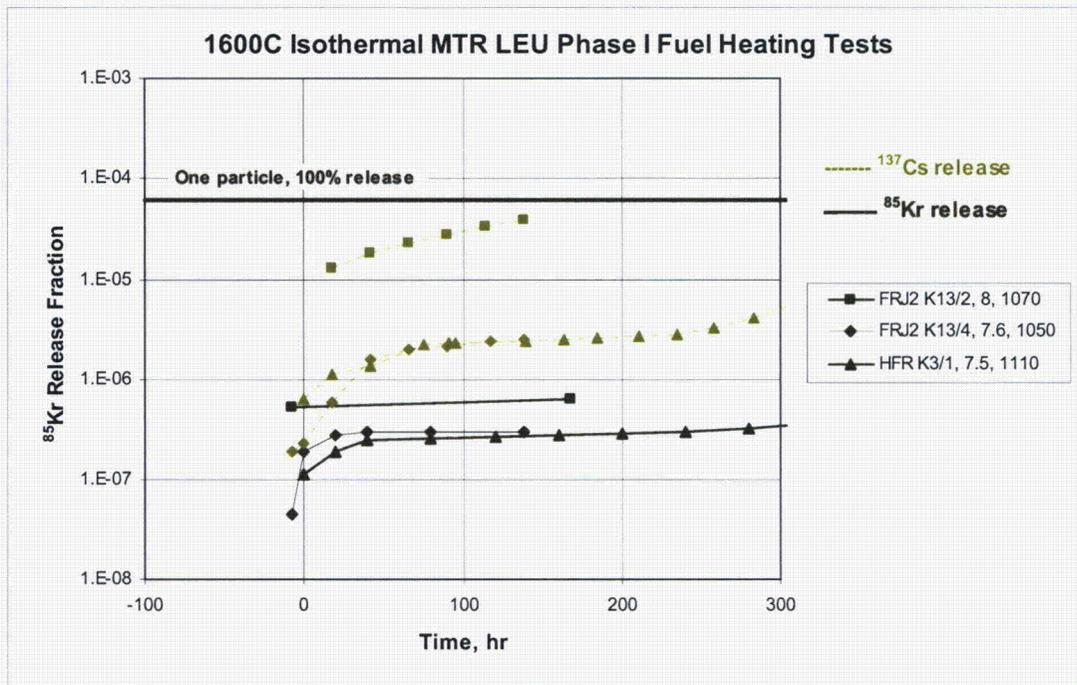


Figure 29: 1600 °C Isothermal LEU Phase I Testing ^{85}Kr and ^{137}Cs Release

3.3.3.3.2 1700 °C isothermal and transient testing

The $^{110\text{m}}\text{Ag}$, ^{90}Sr and ^{137}Cs release data for a selected 1700 °C isothermal heating test are presented in Figure 30 (data from [15]). At 1700 °C, the silver is released more quickly from intact particles, the strontium is retained within the sphere as it was at 1600 °C, and the cesium is retained within the intact particles (as will be discussed in relation to Figure 31, sphere AVR 74/11 had a particle with a defective silicon carbide layer). As discussed in paragraph 3.3.3.2.2, the krypton release indicated a partial particle failure at ~80 h. The cesium release after 80 h changes from an asymptotic approach to a single particle inventory, associated with a particle with a defective SiC layer, to an increasing trend, apparently reflecting release from the partially failed particle.

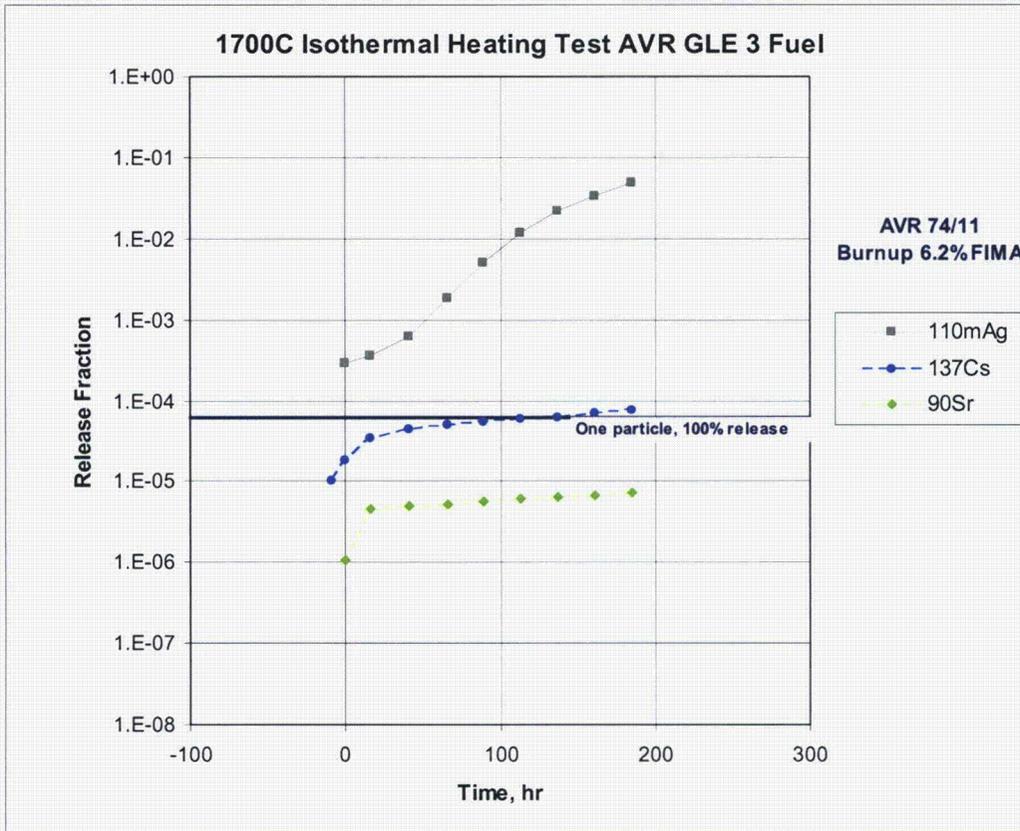


Figure 30: AVR 74/11 1 700 °C Isothermal Testing ^{110m}Ag, ⁹⁰Sr and ¹³⁷Cs Release

The ⁸⁵Kr and ¹³⁷Cs 1 700 °C isothermal and transient heating test results for the AVR GLE 3 fuel are shown in Figure 31. The initial ⁸⁵Kr responses for the 1 700 °C tests indicate there were no through-coating particles failures at the end of irradiation for either of the spheres. The ¹³⁷Cs responses indicate one particle with a silicon carbide defect in sphere AVR 74/11, and no particles with silicon carbide defects in the AVR 91/31 sphere. (The failure of approximately 20 particles beginning approximately 12 h into the transient, as indicated by ⁸⁵Kr, produced the subsequent increase in ¹³⁷Cs release.)

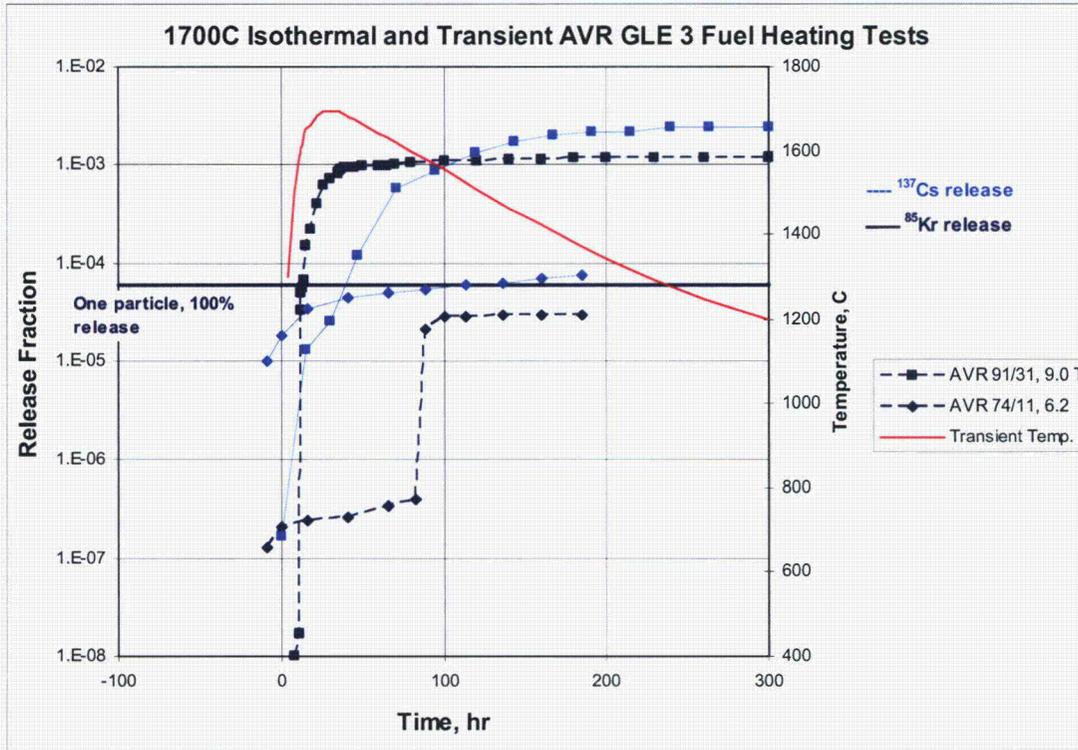


Figure 31: 1 700 °C Isothermal and Transient GLE 3 Testing ⁸⁵Kr and ¹³⁷Cs Release

3.3.3.3.3 1 800 °C isothermal testing

The ^{110m}Ag, ⁹⁰Sr and ¹³⁷Cs release data for a selected 1 800 °C isothermal heating test are presented in Figure 32 (data from [15]). At 1 800 °C, the silver approaches complete release after several hundred hours. Both the cesium and the strontium indicate high levels of release through 'intact' particles. (The krypton release for AVR 76/18 is still only a fraction of a particle inventory at 200 h, as shown in Figure 33, indicating that the pyrocarbon layers are still intact.)

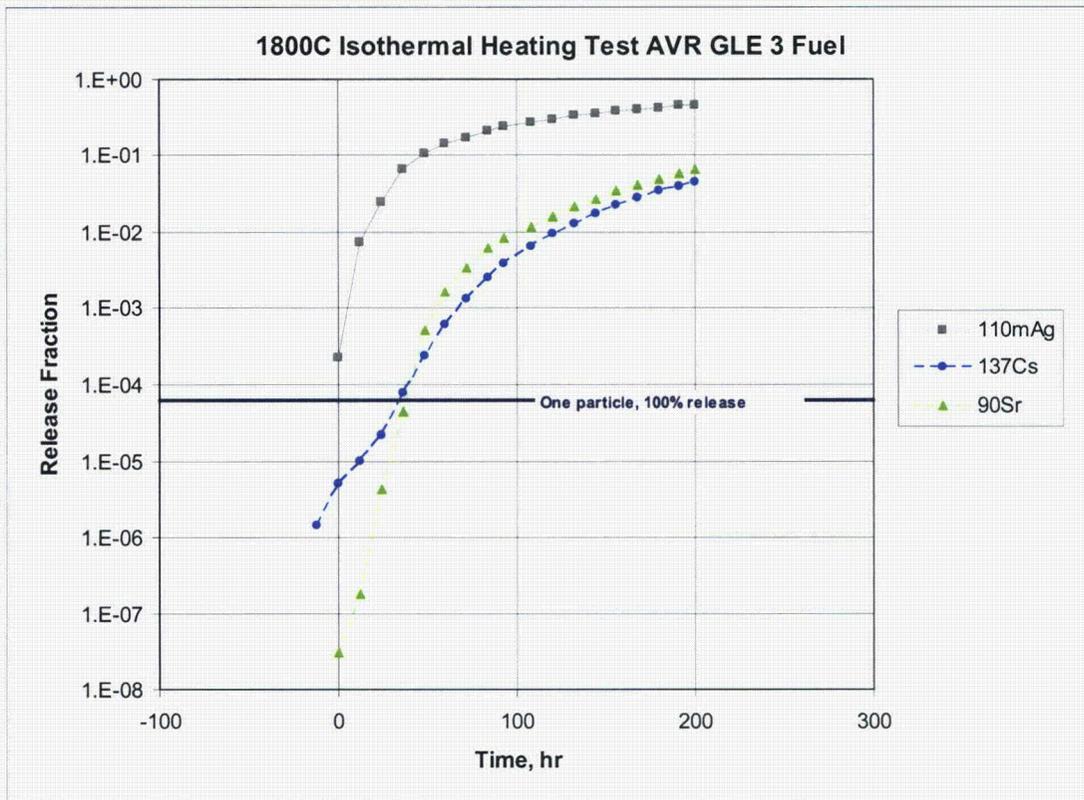


Figure 32: AVR 76/18 1 800 °C Isothermal Testing ^{110m}Ag , ^{90}Sr and ^{137}Cs Release

The ^{85}Kr and ^{137}Cs 1 800 °C isothermal and transient heating test results for the AVR GLE 3 and LEU Phase I fuels are shown in Figure 33 and Figure 34 (data from [15]). These responses require more detailed interpretation than the lower temperatures because the response is faster and the ^{137}Cs is released from intact particles as the test proceeds, as shown in Figure 32. Spheres AVR 88/15 and 88/33 had been shown to be free of through-coating failures by the response to the 1 600 °C heating tests. For the other two GLE 3 spheres and the two LEU Phase I spheres, the initial ^{85}Kr responses for the 1 800 °C tests of the other spheres indicate there were no through-coating particle failures at the end of irradiation. The ^{137}Cs responses to the 1 600 °C heating tests had indicated no silicon carbide defects in AVR 88/15 and three in AVR 88/33, as discussed earlier. The response of the AVR 88/41 sphere was interpreted to indicate two particles with silicon carbide defects at the end of irradiation. The responses for the other GLE 3 sphere and the two LEU Phase I spheres indicate no particles with silicon carbide defects at the end of irradiation.

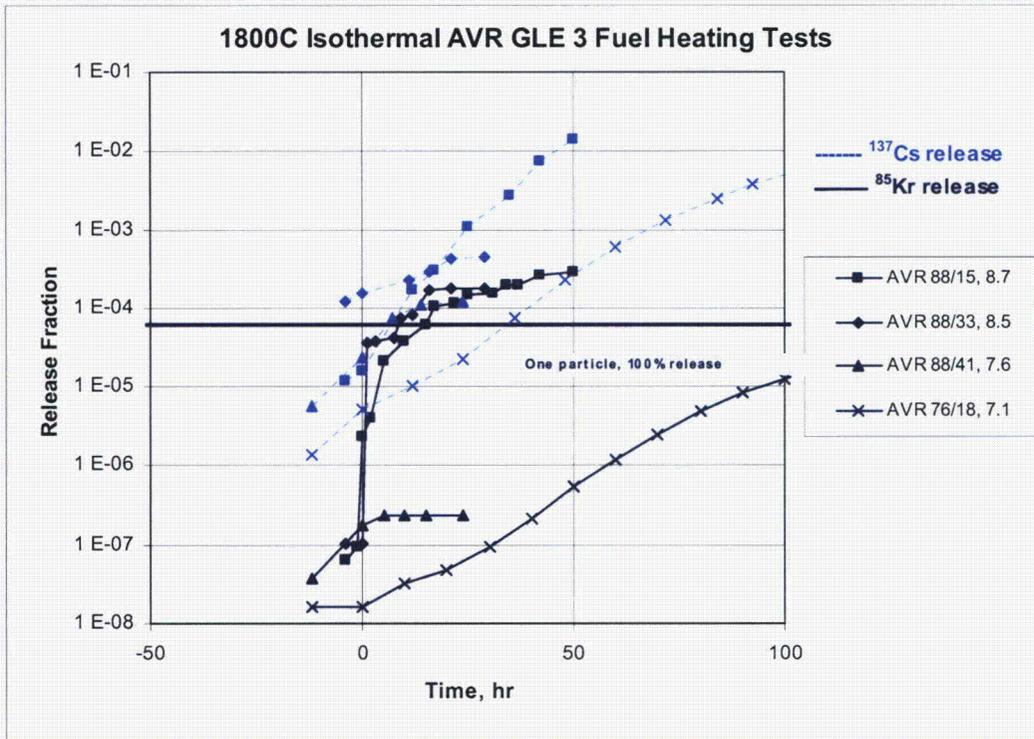


Figure 33: 1 800 °C Isothermal GLE 3 Testing ⁸⁵Kr and ¹³⁷Cs Release

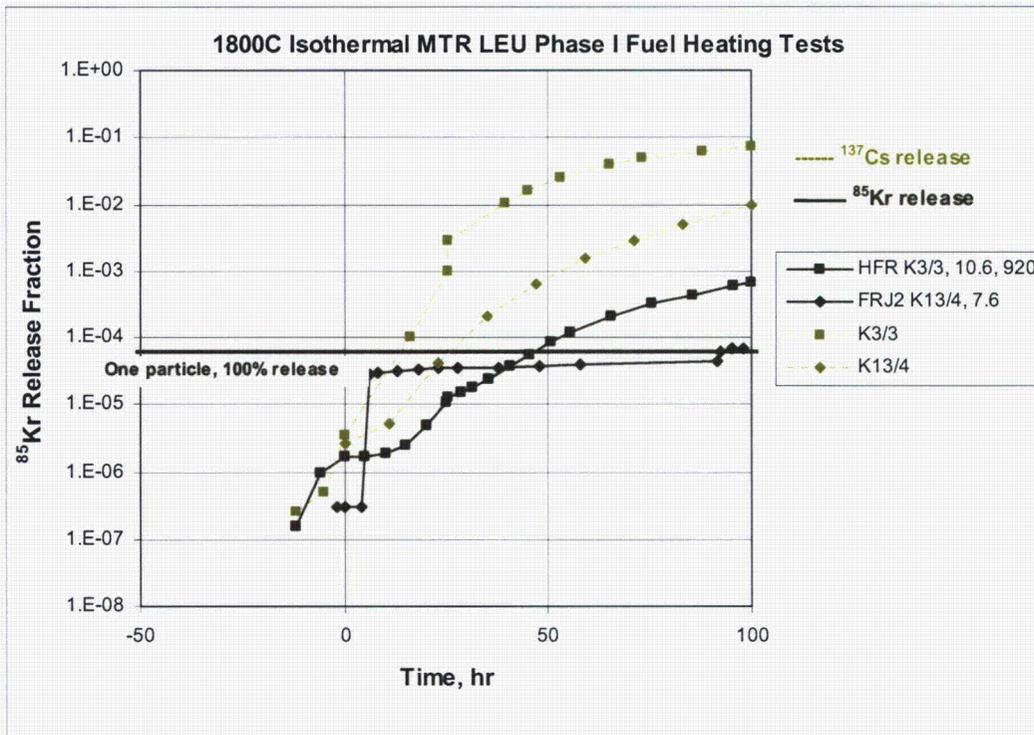


Figure 34: 1 800 °C Isothermal LEU Phase I Testing ⁸⁵Kr and ¹³⁷Cs Release

3.3.3.3.4 Summary of heating test metallic fission product release results

The cesium release response at the beginning of the tests can be used to identify particles with defective silicon carbide layers that were present at the end of irradiation. This is of major value, as the results can be compared with the incidence of silicon carbide defects (as determined by the free uranium fraction) that were detected in the as-manufactured fuel using the burn-leach procedure. The results, as determined by the cesium release analyses of the previous paragraphs, are summarized in Table 13. The implications of these data are discussed further in paragraph 3.3.2.3.

Table 13: End of Irradiation Fuel Condition Inferred from Heating Test Data

ID	Number of Particles	Burnup	Fast Fluence	Irradiation Temperature	Test Temperature	Exposed Kernels	SiC Defects
AVR Spheres*							
AVR 88/15	16 400	8.7	2.4		1 600	0	0
AVR 82/20	16 400	8.6	2.4		1 600	0	1
AVR 88/33	16 400	8.5	2.3		1 600	0	3
AVR 71/22	16 400	3.5	0.5		1 600	0	0
AVR 90/20	16 400	9.8	2.9		1 620T	0	0
AVR 90/2	16 400	9.2	2.7		1 620T	0	1
AVR 90/5	16 400	9.2	2.7		1 620T	0	0
AVR 85/18	16 400	9.15	2.6		1 620T	0	0
AVR 89/13	16 400	9.1	2.6		1 620T	0	0
AVR 74/11	16 400	6.2	1.4		1 700	0	1
AVR 91/31	16 400	9.0	2.6		1 700T	0	0
AVR 88/41	16 400	7.6	1.9		1 800	0	2
AVR 76/18	16 400	7.1	1.7		1 800	0	0
MTR Spheres							
HFR-K3/1	16 400	7.5	4	1 200	1 600	0	0
FRJ2-K13/2	16 400	8	0.2	1 150	1 600	0	1
FRJ2-K13/4	16 400	7.6	0.2	1 120	1 600	0	0
HFR-K3/3	16 400	10.6	5.9	920	1 800	0	0
Total Particles	278800	8.2	2.3	←Average Values Totals→		0	9
50% confidence maximum parent population fraction						2.49E-06	3.47E-05
95% confidence maximum parent population fraction						1.07E-05	5.63E-05
97.5% confidence maximum parent population fraction						1.32E-05	6.13E-05

Note: * Fluence calculated from burnup using correlation from note by Werner (AVR AZ: Hr-X1, 23.5.1984).

The release of silver from intact particles was observed at all heating test temperatures, with the rate of release increasing substantially between 1 600 °C and 1 800 °C. Cesium was seen to be effectively retained in the intact particles at 1 600 °C and for the first 100 h at 1 700 °C, but intact particles dominated the cesium release at 1 800 °C. Strontium was retained within the spheres at 1 600 °C and 1 700 °C, but was released at 1 800 °C.

In the initial 1 700 °C to 1 800 °C heating periods, retention of metallic fission products (exception: silver) is high by comparison with standard prediction methods [7], but the degree of retention is strongly influenced by prior irradiation conditions, notably irradiation temperature, fluence, and burnup.

3.3.4 Summary of German Fuel Design and Experience

The German fuel design adopted for the PBMR evolved from decades of international coated particle fuel fabrication, irradiation, and PIE and testing experience covering a wide range of particle designs, fuel forms, and irradiation and testing conditions. Numerous international bilateral and multilateral data and analytical methods exchanges (e.g., [19]) facilitated the effective incorporation of this experience into the definition and development of the German LEU UO₂ TRISO fuel particle and sphere design that began in the late 1970s. An extensive German fuel development program was conducted through the mid-1990s, resulting in a substantial body of fabrication, irradiation, and PIE and testing data.

The dense UO₂ kernel has proven to be highly retentive of most radionuclides during normal operation, and to a lesser extent at the elevated temperatures associated with some accident conditions. This characteristic must be taken into account when interpreting coated particle release data. The dense pyrocarbon layers in coated particle fuel are highly effective in retention of gaseous radionuclides, but somewhat permeable for metallic radionuclides in normal operation, and more so under accident conditions. The silicon carbide layer was included in the TRISO particle design primarily for its capability to retain metallic radionuclides that had been released from BISO particle designs containing only buffer and dense pyrocarbon coatings around the kernel. Thus two types of coating defects or failures are defined and detectable for TRISO particles:

- **SiC Defects** – Particles with failed or permeable SiC layers but at least one intact dense pyrocarbon layer. These particles will release metallic fission products (e.g., cesium) to a limited extent during normal operation and at a higher level under the elevated temperatures associated with some accident conditions.
- **Exposed Kernels** – Particles with connected failures in both dense pyrocarbon layers and the SiC layers. These particles will release both gaseous and metallic fission products to a limited extent during normal operation and at a higher level under the elevated temperatures associated with some accident conditions.

The burn-leach procedure used to determine the quality of as-manufactured fuel detects both SiC defects and exposed kernels without distinguishing between the two. Gaseous radioisotope release measurements during irradiation in MTRs can be used to identify exposed kernels, but there are no methods to detect SiC defects during the irradiation. However, both exposed kernels and SiC defects in fuel spheres at the end of irradiation in either the AVR or MTRs can be individually identified and distinguished using gaseous and metallic radionuclide release data from the beginning of the post-irradiation heating tests. Distinguishing between these types of coating defects or failures reduces the predicted gaseous radionuclide release during both normal operation and accident conditions, since only the exposed kernel type will release gas.

The German fuel irradiation experience includes both bulk fuel testing in the AVR, and carefully controlled and monitored irradiations in MTRs in Germany, the Netherlands, and France. The irradiation conditions (temperature, burnup, fast fluence) in the MTRs spanned the range of irradiation conditions projected for PBMR fuel in normal operation. The average fuel irradiation conditions in the AVR were more severe than projected for PBMR fuel.

Post-irradiation heatup testing of irradiated German fuel spanned the range of temperature conditions projected for PBMR fuel under accident conditions.

The irradiation temperature environment in the AVR was significantly more severe than projected for the PBMR, thus AVR sphere irradiation data can be used to project performance of fuel in the PBMR. The data and analyses presented and discussed in paragraph 3.3 lead to the following observations and conclusions regarding fuel behavior during normal operation:

- Exposed kernels:
 - In-pile MTR gas release data from eight LEU Phase I and eight Proof Test spheres containing a total of 247 800 particles indicated three particles with exposed kernels at the beginning of irradiation, with no additional failures during irradiation.
 - Post-irradiation heating test krypton release data from 14 GLE 3 spheres containing 229 400 particles irradiated in the AVR indicated no particles with exposed kernels.
- Silicon carbide defects:
 - The nominal as-manufactured free uranium fraction for GLE 3 spheres irradiated in the AVR was 5.07×10^{-5} , and for LEU Phase I spheres irradiated in MTRs it was 3.5×10^{-5} . These results would include both exposed kernels and SiC defects.
 - Post-irradiation heating cesium release data from 13 GLE 3 spheres containing 213 200 particles irradiated in the AVR and four LEU Phase I spheres containing 65 600 particles irradiated in MTRs (a total of 278 800 particles) indicated nine particles with defective SiC layers. This results in a 50% confidence SiC defect fraction of 3.5×10^{-5} and a 95% confidence value of 5.6×10^{-5} .
 - The weighted as-manufactured SiC defect fraction for the 13 GLE 3 spheres and four LEU Phase I spheres discussed above is 4.9×10^{-5} . This is more than the 50% confidence post-irradiation SiC defect fraction (3.5×10^{-5}) and less than the 95% confidence value (5.6×10^{-5}), indicating that no additional SiC defects occurred during irradiation, and that the as-manufactured burn-leach results did not include exposed kernels.
- Both the in-pile gas release (exposed kernels) and the post-irradiation heating cesium release (SiC defects) indicated no particle failures during irradiation in a total of 477 400 particles. These results indicated that particle failures during irradiation are highly unlikely.

The data and analyses presented and discussed in paragraph 3.3 lead to the following observations and conclusions regarding fuel behaviour under accident conditions:

- The heating tests included both isothermal tests at all temperatures and transient simulation tests at 1 600 °C and 1 700 °C. The krypton release data resulted in the maximum parent population exposed kernel fractions at 50%, 95%, and 97.5% confidence, as shown in Table 14.

Table 14: Heating Test Results – Exposed Kernel Fractions

Heating Temperature (°C)		Confidence		
		50%	95%	97.5%
1 600		3.1×10^{-5}	5.6×10^{-5}	6.1×10^{-5}
1 700		6.3×10^{-4}	8.9×10^{-4}	9.4×10^{-4}
1 800	30 h	8.1×10^{-5}	1.3×10^{-4}	1.4×10^{-4}
	50 h	2.7×10^{-4}	3.7×10^{-4}	3.9×10^{-4}
	100 h	8.5×10^{-4}	1.03×10^{-3}	1.06×10^{-3}

- The krypton release results of the heating tests at 1 600 °C and 1 700 °C are dominated by transient simulation data from two AVR spheres at 1 620 °C and one AVR sphere at 1 700 °C. Fuel performance models would predict that the isothermal testing should be more challenging for the fuel, and it is possible that these three spheres experienced exceptionally high temperatures during irradiation in the AVR. It should be noted that three additional spheres were subjected to transient simulation testing at 1 620 °C with no failures. The data at 1 700 °C are limited, including only two spheres. Two of the 1 800 °C tests, which dominate the results, may have seen excessively high temperatures due to fouling of the pyrometer window.
- The metallic radionuclide release results from the heating tests provided a means of identifying particles with SiC defects that were present at the end of irradiation. They also indicated release of silver at all temperatures, with effective retention of cesium and strontium at 1 600 °C and increasing release after extended periods at 1 800 °C.

The heating tests conducted on the German fuel included isothermal tests at temperatures as high as 2 100 °C. While increased particle failure and gaseous fission product release was observed at the elevated temperatures, ⁸⁵Kr release fractions after 25 h at the highest temperature was approximately 1%. Taken as a whole, these data support four important conclusions:

- Substantial margins exist in the ultimate capability of the fuel relative to the PBMR best estimate maximum accident condition temperature of 1 593 °C.
- The degradation in fuel performance at elevated temperatures is regular and gradual. No sudden changes in behaviour (cliff-edge effects) as a function of irradiation temperature, burnup, or accident temperature are seen.
- No exposed kernel particle failures were observed in any of the MTR irradiations.
- The 50% confidence silicon carbide defect fraction after irradiation to substantial burnups in both the AVR and MTRs and heating at 1 600 °C is approximately equal to the mean free uranium fraction of the as-manufactured fuel, indicating no additional SiC defects were produced during irradiation.

Taken as a whole, the body of data produced in the German heatup testing includes fuels that span the range of burnup, fast fluence, and temperature expected for the PBMR, and that were irradiated in both the prototypical cyclical environment of the PBMR core and in isothermal conditions. The heatup testing includes both isothermal conditions and time-temperature profiles that approximate a PBMR DLOFC transient. The data are generally consistent and coherent with regard to irradiation temperature, burnup, fluence, and heating test temperature, notwithstanding the statistical limitations of dealing with very low probabilities of particle failures, the random nature of outlier particle defects (e.g., silicon carbide layer flaws, faceting) and characteristics (e.g., combinations of kernel diameter and

layer thicknesses), and the potential for failure modes with varying ^{85}Kr release characteristics.

While the German LEU UO_2 TRISO fuel irradiation and heating test database provide a firm basis for projecting performance of PBMR fuel, some areas will be subject to further work to support operation of the PBMR:

- Data will be generated to provide assurance that the PBMR fuel is sufficiently similar and its performance is consistent with the performance of the German fuel, and to demonstrate 'equivalence.'
- The existing data at 1 600 °C and 1 700 °C suggest that the transient temperature profile could be more limiting than the isothermal test. The statistics are limited and analytical models indicate that the isothermal test should be more challenging to the fuel. This area may be addressed analytically or by conducting additional transient simulation tests.
- The existing data at 1 700 °C are sparse and not consistent with the expected exponential dependence of particle failure on temperature, relative to the 1 600 °C and 1 800 °C data. Additional data will be required to provide sufficient confidence in the fuel performance at 1 700 °C.
- The existing data at 1 800 °C are dominated by the results of tests conducted in the 'A'-test furnace that may have experienced temperatures significantly above the recorded temperature due to fouling of the pyrometer window. Additional data will provide greater confidence regarding the actual failure fraction at this temperature.

The planned PBMR fuel irradiation and testing program discussed in paragraph 3.4 is intended to address these areas.

3.4 PBMR FUEL QUALIFICATION PROGRAM

The PBMR approach to fuel manufacture is to ensure that the manufacturing process is equivalent to that used for manufacture of German LEU UO_2 TRISO fuel. However, some additional fuel performance data will be gathered to confirm equivalence to the German fuel has been achieved, and will add to the database for application to the PBMR fuel service conditions. Therefore, additional irradiation and post-irradiation testing and examination will be performed on PBMR-manufactured fuel. This paragraph summarizes the planned PBMR fuel test program, PIEs, and plant startup surveillance.

The PBMR fuel design is based on the German LEU UO_2 TRISO design for which an extensive experience base exists as discussed in paragraph 3.3.4. This experience base consists of mechanical, irradiation, and post-irradiation heating tests performed on fuel produced in Germany in the period from 1981 to 1988. The German fuel development program culminated in state-of-the-art fuel manufactured in 1985 for bulk testing in the AVR (AVR 21-2 reload), and in 1988 for the HTR-Modul Proof Tests. As PBMR uses both the German Proof Test fuel design and a manufacturing process equivalent to that for German fuel, it is not necessary to repeat all the irradiation tests from which the operating envelope for the German LEU UO_2 TRISO fuel was derived. PBMR, however, plans to expand the existing database. The additional data will (1) demonstrate successful replication of the German design in PBMR manufacturing facilities, (2) cover the full range of PBMR design and operating conditions and requirements, (3) strengthen the statistical confidence of the performance base for PBMR fuel, and (4) reaffirm expected in-reactor performance under PBMR operating conditions. A comparison of the PBMR operating envelope to the German LEU UO_2 TRISO irradiation and heating test experience was presented in paragraph 3.3.2.

The PBMR fuel test program can be considered to be an evolving program in that it will be revised to address new test requirements, revised analyses, or new understandings of fuel performance based on early test results as new knowledge is gained. Values provided in the following paragraphs are nominal or best-estimate and may be revised as the particular test is specified in more detail prior to its initiation.

3.4.1 Materials Test Reactors

The PBMR fuel irradiation and testing program will be conducted in the following facilities:

- **IVV-2M** – The test reactor planned to be used for the PBMR production plant fuel irradiation program and a portion of the PBMR laboratory produced fuel irradiation program is the IVV-2M reactor located at Zarechny in the Russian Federation [20]. The IVV-2M reactor consists of a water-moderated reactor core with a nominal power output of 15 MWt, surrounded by a beryllium reflector. The reactor is operated in 300 h cycles, with shutdowns of approximately two days between cycles, and is shut down twice yearly for refuelling. Fuel sphere irradiation tests are performed in irradiation rigs that can accommodate four full-sized fuel spheres per rig.
- **HFR** – A portion of the PBMR laboratory produced fuel irradiation program will be conducted in the HFR reactor located at Petten in the Netherlands [21]. The HFR consists of a water-moderated reactor core with a nominal power of 45 MWt, surrounded by a beryllium reflector. The reactor is operated on a nominal annual schedule consisting of 10 to 11 ~28 day cycles with two shutdowns of several weeks for maintenance and related activities. Fuel sphere irradiation tests are performed in irradiation rigs with multiple independent capsules and up to a total of five full-sized fuel spheres.

3.4.2 Testing of Laboratory-Produced Fuel Spheres

A total of nine 'pre-production' fuel spheres manufactured in the PBMR fuel laboratory will be irradiated to provide early confirmation of the validity of the PBMR manufacturing process. Five spheres will be irradiated in the Petten reactor in the Netherlands, and four will be irradiated in the IVV-2M reactor in the Russian Federation. The fuel spheres for these tests will contain coated particles produced in a full-sized coater prototypical of the coaters to be installed in the PBMR pilot fuel plant. The results of these tests will be used to confirm the performance of fuel produced in the full-scale pre-production or 'advance' coater as part of the manufacturing assurance program.

3.4.3 Testing of Production Plant Fuel Spheres – Overview

A pilot fuel plant will be constructed in South Africa to produce fuel in sufficient quantities to support the startup and continued operation of the PBMR demonstration plant. This portion of the testing is on equilibrium fuel spheres manufactured on a fully qualified production line from the pilot fuel plant. PBMR will be started up with fuel enriched to approximately 4.2% to 4.5%⁴ ²³⁵U to control startup reactivity, and will then be transitioned to the use of equilibrium fuel whose ²³⁵U enrichment will be approximately 9.6%. The first test, using four equilibrium fuel spheres, will be ended when the spheres reach a burnup of approximately 5% FIMA (which corresponds to the maximum expected burnup of startup fuel in PBMR). In the second test, a total of 12 equilibrium fuel spheres will be irradiated until the maximum burnup of

⁴ The enrichment of the startup fuel is expected to be in this range, with a final determination to be made following more detailed analysis of the startup and initial operation of the core.

approximately 11.6% FIMA is reached. Additional detail on these tests is provided in paragraphs 3.4.4 and 3.4.5.

3.4.3.1 Testing objectives

The fuel spheres manufactured on a qualified PBMR production line will meet all specified requirements and this will be confirmed independently during pre-irradiation characterization. Tests will confirm that the behaviour of PBMR fuel spheres under irradiation and post irradiation heating test conditions meets PBMR requirements for normal operating and accident conditions regarding:

- Fission gas release – Fission gas release will be measured in an external gas loop during irradiation and during heating tests.
- Metallic fission product release – Metallic fission product releases that occur during irradiation will be determined in the PIE by measuring deposition in the irradiation capsules that surround the fuel spheres during irradiation. Releases will be periodically determined during heating tests by removing and replacing deposition plates in the test furnace.
- Failure of coated particles – Exposed kernel coated particle failures that may occur during irradiation and during heating tests can be quantified by means of fission gas release analysis.

3.4.3.2 Pre-irradiation characterization

Fifteen fuel spheres, in addition to the 16 to be irradiated, will be produced for independent pre-characterization. This will form part of an independent assessment of the as-manufactured properties of coated particles, fuel spheres and matrix graphite. Pre-irradiation characterization for fuel spheres and coated particles will consist of:

- Determination of geometrical sizes of kernels, coated particles, and coating layer thickness.
- Determination of kernel density, coated particle density, and density of all coating layers.
- Kernel and coated particle mass determination.
- Determination of kernel impurities.
- Ceramography of coated particles to:
 - Investigate the structure of the kernel, buffer, PyC, and SiC layers.
 - Determine anisotropy of dense PyC layers, including in particles removed from a heat treated fuel sphere.
 - Determine uranium distribution in coated particle layers.
- Determination of free uranium content in spheres.
- Determination of U/O ratio in kernels.
- Determination of compression strength of coated particles.

3.4.4 Testing of Production Plant Fuel Spheres – Partial Burnup

The purpose of this test is to confirm that fuel manufactured for use in the PBMR demonstration power plant can be burned to approximately 5% FIMA, the maximum burnup calculated for the startup fuel, and will perform satisfactorily under DBA conditions. Four equilibrium fuel spheres containing a total of approximately 57 800 coated particles will be

loaded into an irradiation rig and inserted into the IVV-2M test reactor. At a burnup of nominally 5% FIMA, the fuel spheres will be removed from the test reactor and subjected to PIE and heating tests. The successful completion of this test will confirm production fuel integrity for normal operation and DBA conditions to a burnup of 5% FIMA.

3.4.4.1 Irradiation targets

Production fuel in the irradiation rig will be irradiated to target burnup values of approximately 5% FIMA (48,200 MWd/t) at a constant center temperature of 1 200 °C. The fast neutron dose at this burnup value is approximately $1.7 \times 10^{21} \text{ cm}^{-2}$. The duration of this irradiation will be roughly one year.

3.4.4.2 Post-irradiation examination

The following PIEs will be performed on the four fuel spheres from the irradiation rig:

- Appearance
- Mass
- Diameter
- Burnup
- Fission product inventory

All irradiated fuel spheres will be subjected to heating tests simulating DBA transient temperatures, first at 1 600 °C for 100 h and then at 1 800 °C for 100 h. Following the heating tests, all heated fuel spheres will be visually examined and their fission product inventories measured.

One heated fuel sphere will then be deconsolidated to provide coated particles for ceramography and fission product distribution measurements including the following.

- Fission product distribution in fuel sphere.
- Optical ceramography of coated particles.
- Irradiated Microsphere Gamma Analysis (IMGA) on coated particles.
- Fission product distribution in coated particles.

3.4.5 Testing of Production Plant Fuel Spheres – Full Burnup Proof Test

The purpose of the proof test is to demonstrate, using a test sample that is statistically sufficient, that production fuel spheres meet all coated particle failure and fission product release requirements under normal operation, anticipated operational occurrences, and DBA conditions, including DLOFC and PLOFC temperature transients. It is planned to irradiate 12 production fuel spheres containing a total of approximately 173 400 coated particles in the IVV-2M reactor.

3.4.5.1 Irradiation target

The average target burnup for PBMR fuel spheres is approximately 9.6% FIMA, which was calculated based on six cycles through the reactor core. In PBMR operation, a small number of fuel spheres could be recycled near the maximum Burnup Measurement System setpoint with uncertainty, and pass through the high burnup path near the inner reflector. Therefore, depending on the setpoint of the Burnup Measurement System and the measurement

uncertainty, a fuel sphere could achieve a burnup as high as approximately 11.2% FIMA (109 000 MWd/t). For this reason, the value of 11.6% FIMA is listed in Table 16 (test summary). The maximum fast neutron dose estimated in a similar manner is approximately $3.6 \times 10^{21} \text{ cm}^{-2}$. Normal temperature cycles in the reactor core will be simulated by irradiation at two representative core temperatures – one interval at a temperature representative of the low-temperature part of the PBMR temperature cycle and a second interval at a temperature representative of the high-temperature part of the PBMR temperature cycle. At the end of irradiation, the fuel spheres in one of the irradiation rigs will be subjected to a temperature transient simulating the first part of a PLOFC transient.

In Table 15, nominal irradiation targets are compared to parameter values.

Table 15: Comparison of Nominal Proof Test Irradiation Targets with Core Design Parameters

Parameter	Design Limits	Design	Nominal Proof Test Irradiation Target
Number of Cycles	6	6	8
Maximum power per fuel sphere (kW)	4.5	2.76	3
Average residence time (days)	-	925	731
End of life fast neutron dose ($E > 0.1 \text{ MeV}$)(cm^{-2})	-	2.72×10^{21}	3.63×10^{21}
Average discharge burnup (MWd/t/% FIMA)	92 000/9.5	91 000/9.4	111,900/11.6
Maximum Temperature (°C):			
Normal operation	1 130	1 068	900/1 150 cycles
DLOFC (transient peak)	-	1 593	1 600/1 800
PLOFC (transient peak)	-	1 319	1 350

3.4.5.2 Post-irradiation examination

The following PIEs will be performed on one or more of the 12 irradiated fuel spheres:

- Appearance
- Mass
- Diameter
- Burnup
- Fission product inventory
- Deconsolidation
- Fission product distribution in fuel sphere
- Optical ceramography of coated particles
- IMGA on coated particles
- Fission product distribution in coated particles

Following irradiation, irradiated fuel spheres will be externally examined and their burnup measured. One fuel sphere will be deconsolidated to enable ceramography of coated particles to be carried out and to measure the fission product distribution through the fuel sphere.

Of the remaining 11 fuel spheres, five spheres will be subjected to heating tests simulating maximum reactor fault transient temperatures, nominally 1 600 °C for about 100 h, and six

spheres will be subjected to temperatures of approximately 1 800 °C for about 100 h. Following heating tests all heated fuel spheres will be visually examined and their fission product inventories measured.

A summary of the planned irradiation tests for PBMR production-plant fuel spheres is provided in Table 16.

3.4.6 Fuel Spheres from PBMR Operation

The German design basis for the PBMR fuel includes conservative design requirements on graphite abrasion resistance, drop-test integrity, and crush-test integrity. The appropriateness of these requirements with regard to assuring mechanical integrity during in-reactor operation was addressed via German laboratory tests as well as tests on fuel spheres that had been irradiated in the AVR. During the startup and initial operation of the PBMR demonstration plant, observations and tests will confirm the German design bases for mechanical integrity. The integrity of the fuel spheres removed from PBMR will be monitored during normal operation by (1) the Burnup Measurement System and by (2) continuously monitoring the fission gas concentration in the reactor helium. In addition, irradiated fuel spheres will be removed from PBMR and subjected to mechanical tests and PIE. This post-irradiation mechanical test and inspection program will be continued until the target burnup for the first equilibrium fuel load is reached.

3.4.7 Quality Assurance Program

The PBMR Quality Assurance Program will state the quality requirements for all aspects of the design, manufacture, and testing of PBMR fuel. The fuel plant will be designed, built, and commissioned in accordance with the quality standard ISO 9000. Fuel design and manufacture will comply with the requirements of ASME NQA-1 [22].

The pilot fuel plant Quality Assurance Program will specify quality assurance requirements for each of the components of the overall program, such as the manufacture and delivery of nuclear fuel products by the pilot fuel plant and any relevant subcontractors. Each component of the overall program will establish appropriate control arrangements to ensure that the quality arrangements are fully implemented. A specific fuel plant Quality Management Program will be established to ensure adherence to the quality assurance requirements for fuel manufacture and materials procurement. In the course of the design certification review, a description of the PBMR quality assurance program will be submitted to show that the PBMR program meets the requirements of 10 CFR 50 Appendix B.

3.4.8 Summary of PBMR Fuel Qualification Program

The PBMR fuel qualification program is directed toward augmenting the existing irradiation and testing data for German LEU UO₂ TRISO fuel through the irradiation and PIE and testing of fuel manufactured by PBMR. The objectives of the program are to:

- Demonstrate successful replication of the German design in PBMR manufacturing facilities.
- Cover the full range of PBMR design and operating conditions and requirements.
- Strengthen the statistical confidence of the performance base for PBMR fuel.
- Reaffirm expected in-reactor performance under PBMR operating conditions.

Current plans for irradiation and testing were discussed in the preceding paragraphs with the understanding that adjustments in the program may occur as more information becomes available.

Irradiation and testing of 'pre-production' fuel fabricated in a full-sized prototypical coater will be conducted in both the HFR reactor in the Netherlands and the IVV-2M reactor in Russia. The results of these tests will provide early confirmation of successful replication of the German fuel with comparable performance. Subsequent irradiation and testing will be conducted in the IVV-2M reactor on equilibrium fuel spheres from the pilot fuel plant, which will produce fuel for the startup and continued operation of the PBMR demonstration plant. The data produced will include pre-irradiation characterization, in-pile gas release, and PIE and safety testing data. Production fuel irradiation and testing will be conducted in two phases:

- **Partial Burnup** – This test will consist of a single irradiation rig containing four spheres and taken to a burnup of approximately 5% FIMA. The resulting data will be used to support startup and initial operation of the PBMR demonstration plant.
- **Full Burnup** – These tests will consist of three irradiation rigs in the IVV-2M reactor, each containing four spheres. The irradiations will be conducted to levels of fast fluence and burnup that exceed the maximum values achievable during PBMR operation.

The data produced by the successful completion of the MTR irradiation and testing of PBMR fuel, in combination with the existing data on German LEU UO₂ TRISO fuel, will demonstrate that PBMR production fuel spheres meet all coated particle failure and fission product release requirements under normal operation, anticipated operational occurrences, and DBA conditions.

Additional confirmatory data under actual PBMR service conditions will be provided by a post-irradiation test and inspection program to be conducted on fuel discharged from the PBMR demonstration plant. This program will be continued until the target burnup for the first equilibrium fuel load is reached.

Table 16: Summary of PBMR Fuel Irradiation Tests

Test	Fuel to be Tested		Test Description						Technical Objective
	Production Route	No.	Test Reactor	Burnup (% FIMA)	Temperature (°C)	Cycle	Post Heat	PIE	
Production Plant Fuel – Partial Burnup Demonstration	Pilot Fuel Plant	4	IVV-2M	5	1 200	No	Four fuel spheres to 1 600 °C, then to 1 800 °C	Y	Qualification to 5% FIMA. Zero or low number of coated particle failures.
Production Plant Fuel – Full Burnup Demonstration	Pilot Fuel Plant	12	IVV-2M	11.6	900/1 150	Yes	Eleven fuel spheres, five to 1 600 °C, six to 1 800 °C	Y	Full fuel proof test, including simulated PLOFC in test reactor. Detailed PIE.

4. ISSUES FOR PRE-APPLICATION RESOLUTION

The purpose of Chapter 4 is to summarize the issues for which PBMR requests feedback during the pre-application review. For this paper, the primary issue is whether or not the subset of German data selected for the validation of the PBMR fuel source term analysis approach, combined with the additional tests to be performed by PBMR, is adequate for that purpose. The following paragraphs present several separate sub-issues, all of which are related to this primary issue.

4.1 ADEQUACY OF THE SUBSETS OF GERMAN DATA SELECTED FOR PBMR

Three subsets of the German experience database are of special interest to PBMR: manufacturing, normal operation irradiation, and transient heatup. The manufacturing experience data subset is discussed in paragraph 3.3.1. Manufacturing data to be applied to PBMR fuel analysis will be restricted to the 'post-1985' manufacturing process adopted by PBMR. Fuel manufactured with that process showed improved burn-leach test results relative to those from the pre-1985 manufacturing process, and provides greater assurance of acceptable performance.

The 'normal operation' experience subsets are described and analyzed in paragraphs 3.3.2 and 3.3.3, and listed in Table 8. The data sets were limited to full-size spheres of similar enrichment and particle loading as the PBMR fuel specification, including fuels irradiated in both the AVR and MTRs. The number of particles with exposed kernels was determined by the initial ^{85}Kr release response in post-irradiation heating tests for spheres irradiated in the AVR, and by in-pile gaseous fission product release for spheres irradiated in MTRs. The number of particles with silicon carbide defects was determined by the initial ^{137}Cs release response in post-irradiation heating tests for spheres irradiated in both the AVR and MTRs. This resulted in a total population of 30 spheres (477 400 particles) for exposed kernel analysis and 17 spheres (278 800 particles) for silicon carbide defect analysis, as given in Table 8. The PBMR fuel performance analysis methods use the core average failure fraction as input, calculating releases based on a spatial integration of fuel operating conditions. The PBMR core average conditions (burnup, fast fluence, and temperature) are substantially less severe than the average for the populations of spheres considered in this analysis. The substantial majority of fuel spheres in both sets were from the GLE 3 and LEU Phase I manufacturing campaigns, which both had significantly higher as-manufactured free uranium fractions than the Proof Test fuel, which is the basis for PBMR fuel manufacturing. Thus the results obtained from the selected data sets can be expected to significantly over predict the silicon carbide defects in the PBMR fuel (no exposed kernels were identified in any of the spheres).

The transient heatup fuel sphere subsets used for determining failure fractions at 1 600 °C, 1 700 °C, and 1 800 °C are listed in Table 9, Table 10, and Table 11 respectively, and the overall results are summarized in Table 12.

While PBMR believes that the above subsets were selected with an appropriate level of conservatism, NRC general concurrence and/or comments are requested.

4.2 ADEQUACY OF PBMR TEST PROGRAM

The PBMR test program (paragraph 3.4) includes normal operation and heatup testing of fuel manufactured on a qualified PBMR manufacturing line. While the test parameters have not yet been finalized, the testing of the 12 spheres at conditions which would meet or exceed PBMR limiting operating conditions would demonstrate performance comparable to that of

the German fuel, and strengthen the normal operation database. Similarly, the subsequent heatup testing of 11 of those spheres would demonstrate performance comparable to that of the German fuel, and strengthen the accident condition database.

PBMR requests NRC general concurrence and/or comments on the adequacy of the German data subsets mentioned in this paragraph, and on the statistical strengthening of those data subsets by the expected tests of PBMR-manufactured fuel.

4.3 DATA QUALIFICATION

The German fuel manufacturing, irradiation, and testing were conducted with a high level of professionalism and attention to considerations that affected the quality of the materials and data produced. The success of their efforts is reflected in a high level of quality in the materials produced and of consistency in the large body of data produced during irradiation as well as PIE and testing. However, the data were not produced under a quality assurance program that explicitly met the requirements of 10CFR50 Appendix B. The utilization of the selected German data identified in this document is essential to the effective analysis of PBMR safety performance.

In addition, the fuel manufacturing, irradiation, and PIE and testing program to be conducted by PBMR and participating organizations will produce data essential to the PBMR design certification review. These activities will be conducted under quality assurance programs compliant with ISO 9000, and the fuel design and manufacturing will comply with ASME NQA-1.

PBMR requests NRC comments on the activities necessary to assure that the German and PBMR data are qualified for use in the design certification application.

5. PRE-APPLICATION OUTCOME OBJECTIVES

For the issues summarized in Chapter 4, it is requested that NRC reviewers provide either:

- confirmation that PBMR plans for addressing the issues are generally acceptable; or
- identification of any additional information needs of the NRC or any areas in which the NRC believes that plans of PBMR will not be sufficient to address applicable regulatory requirements and guidance.

To facilitate NRC review of the papers and issuance of RAIs, it is expected that a workshop will be held following initial NRC review of the paper.

It is then proposed that RAIs be issued and that a second workshop be conducted to discuss the RAIs.

Following the second workshop, PBMR expects to revise this paper to respond to the RAIs to the extent feasible during the pre-application review period.

Finally, it is expected that the revised paper would be submitted and that the NRC would issue its closure documentation.

In addition, it is expected that, during the workshops and in the closure documentation, NRC staff would identify any policy issues for which Commission direction is required.

6. APPENDICES

6.1 APPENDIX A: GERMAN FUEL IRRADIATION PERFORMANCE DATA FROM AVR

As noted earlier, these data are compiled from [9], [13], [14], and [15]. Table 13 in paragraph 3.3.3.3.4 lists exposed kernels and silicon carbide defects in irradiated AVR 19 (GLE 3) spheres that were inferred from the initial response to heating tests.

In addition to the GLE 3 spheres irradiated in AVR that were subjected to heating tests as discussed in paragraph 3.3.3, GLE 3 and GLE 4 spheres were destructively examined to determine the fission product profile in the fuel-free zone, as discussed in section 7.2.3 of [15]. In this procedure, the fuel-free zone was mechanically turned off in steps and the samples were examined with gamma spectrometry to determine the concentration profile of fission products in the fuel-free zone. The cesium profiles for spheres with no silicon carbide defects decreased from the outside surface toward the fuelled region, reflecting exterior contamination collected as the cooled spheres were in the extraction tube. A sphere containing a particle that was releasing cesium produced a profile that turned upward moving into the fuel free zone. Some representative profiles, including a sphere with a releasing particle, are provided in Figure 35.

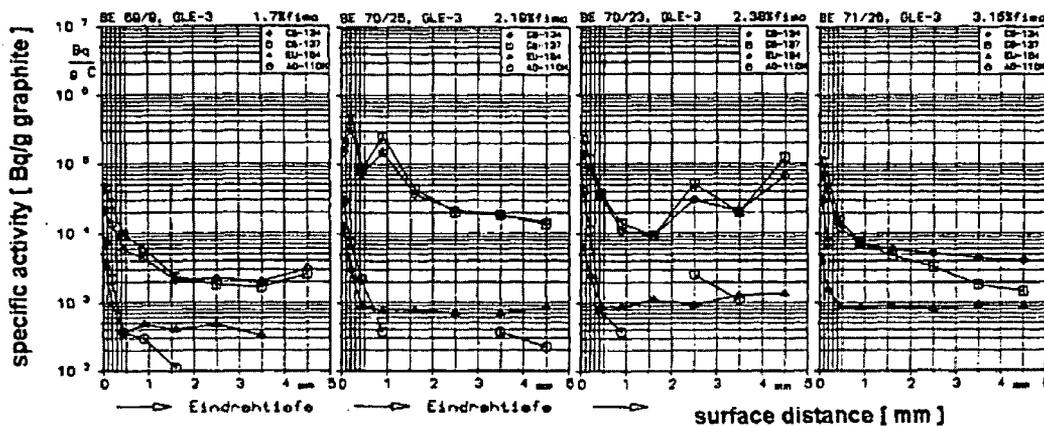


Figure 35: Representative Fission Product Profiles in Fuel-free Zone [15]

The results of the fission product profile examinations of fuels irradiated in the AVR are provided in Table 17. This procedure was indeterminate as to whether the particle releasing fission products was a particle with an exposed kernel or with a silicon carbide defect with an intact pyrocarbon layer. The heating test data would strongly indicated that particles releasing cesium are particles with silicon carbide defects, but since the procedure cannot make the distinction, the exposed kernel column indicates 'N/D' for not determined, and the sphere is not included in the statistics for exposed kernels. The fast fluence values of the spheres were estimated by a correlation as noted at the bottom of the table. This is a nominal value, with the actual value for a given sphere determined by the trajectory taken on passes through the core. The variation in fluence due to this effect was estimated to be less than 10%.

Table 17: GLE 3 and GLE 4 Sphere Failure Fractions from Fission Product Profile in Fuel-free Zone

Sphere	No. of Particles	Burnup	Fast Fluence ¹	Exposed Kernels	SiC Defects ²
GLE 3 Spheres					
69/9	16 400	1.7	0.19	0	0
70/25	16 400	2.2	0.24	0	0
70/23	16 400	2.4	0.27	N/D	1
71/26	16 400	3.2	0.41	0	0
71/19	16 400	3.9	0.58	0	0
73/31	16 400	5.9	1.23	0	0
76/32	16 400	6.4	1.43	0	0
76/30	16 400	6.7	1.55	0	0
80/19	16 400	7.2	1.76	0	0
76/33	16 400	7.7	1.98	0	0
80/33	16 400	8	2.11	0	0
80/32	16 400	8.8	2.47	0	0
81/22	16 400	9.1	2.61	0	0
GLE 4 Spheres					
77/11	9 560	4	0.36	0	0
77/2	9 560	4.5	0.42	0	0
77/7	9 560	4.7	0.44	0	0
80/7	9 560	7.5	0.86	0	0
79/4	9 560	8.3	1.02	0	0
82/4	9 560	9.2	1.22	0	0

Notes:

1. Fluence calculated from burnup using correlation from note by Werner (AVR AZ: Hr-X1, 23.5.1984).
2. Failure fraction determined by ¹³⁷Cs profile.

Five additional GLE 4 spheres were subjected to deconsolidation for further examination of individual particles. This process supported identification of failed particles by measuring the fission product concentration in the leachate as the sphere was progressively deconsolidated. The results are interpreted assuming that uranium in the leachate would be from a particle with an exposed kernel, and should be included in both the exposed kernel and silicon carbide defect columns. It is conceivable that the particle was damaged in the deconsolidation process, but it will be included as a failed particle from manufacturing or irradiation. This interpretation produces the results given in Table 18.

Table 18: GLE 4 Sphere Failure Fractions from Deconsolidation Leachate

Sphere	No. of Particles	Burnup	Fluence ¹	Exposed Kernels	SiC Defects ²
78/1	9 560	3.5	0.32	0	0
78/2	9 560	3.5	0.32	0	0
78/3	9 560	3.5	0.32	1	1
78/4	9 560	3.5	0.32	0	0
78/5	9 560	3.5	0.32	0	0

Notes:

1. Fluence calculated from burnup using correlation from note by Werner (AVR AZ: Hr-X1, 23.5.1984).
2. Failed particles identified by progressive deconsolidation and examination of leachate.

The results listed above from PIE of GLE 3 and GLE 4 fuel spheres irradiated in the AVR are included with the exposed kernel and silicon carbide defect fractions inferred by initial response during heating tests in the evaluation of normal operation fuel performance in paragraph 3.3.2.

6.2 APPENDIX B: GERMAN FUEL IRRADIATIONS IN MATERIALS TEST REACTORS

The data from MTR irradiations of LEU UO₂ TRISO fuel in the following tables were drawn from [12], [13], [14], [15], and [7]. Reference [23] lists the German fuel specimens that were irradiated in MTRs in the Netherlands and France in locations that had harder spectra than PBMR, and thus produced higher fast fluences. The HFR-P4 and SL-P1 specimens were small (~2 cm diameter) fuelled spheres embedded in a matrix cylinder. The HFR Kx specimens were full-sized spheres from the LEU Phase 1 (HFR K3) and Proof Test (HFR K5 and K6) campaigns (refer to Table 4). The temperatures are sphere center values consistent with the PBMR service conditions of Table 2.

Table 19: High Fluence Irradiations in the Netherlands (HFR) and France (SL)

Experiment	Capsule	Compact or Sphere*	Number of Particles	Burnup	Irradiation Temperature	Fast Fluence	Heating Temperature
HFR-P4	1	1	1 631	11.6	999	7	
HFR-P4	1	2	1 631	12.4	982	7.3	
HFR-P4	1	3	1 631	12.3	965	7.6	
HFR-P4	1	4	1 631	12.5	953	7.8	
HFR-P4	1	5	1 631	12.5	941	8	
HFR-P4	1	6	1 631	13.5	941	7.9	
HFR-P4	1	7	1 631	12.6	940	7.5	
HFR-P4	1	8	1 631	12.1	973	7.2	1 600
HFR-P4	1	9	1 631	12.1	1 006	7	
HFR-P4	1	10	1 631	10.2	1 007	6.5	
HFR-P4	1	11	1 631	10.4	1 008	6.1	
HFR-P4	1	12	1 631	9.7	1 008	5.5	1 600
HFR-P4	3	1	1 631	11.0	1 010	7	
HFR-P4	3	2	1 631	11.3	1 020	7.3	
HFR-P4	3	3	1 631	11.7	1 030	7.6	
HFR-P4	3	4	1 631	12.9	1 023	7.8	

Experiment	Capsule	Compact or Sphere*	Number of Particles	Burnup	Irradiation Temperature	Fast Fluence	Heating Temperature
HFR-P4	3	5	1 631	12.3	1 015	8	
HFR-P4	3	6	1 631	12.3	1 017	7.9	
HFR-P4	3	7	1 631	12.2	1 019	7.5	1 600
HFR-P4	3	8	1 631	11.5	1 051	7.2	
HFR-P4	3	9	1 631	11.1	1 082	7	
HFR-P4	3	10	1 631	11.0	1 082	6.5	
HFR-P4	3	11	1 631	9.6	1 082	6.1	
HFR-P4	3	12	1 631	8.7	1 082	5.5	1 800
SL-P1		1	1 666	7.5	743	5	
SL-P1		2	1 666	8.1	750	5.4	
SL-P1		3	1 666	8.8	759	5.8	
SL-P1		4	1 666	9.3	785	6.2	
SL-P1		5	1 666	9.6	788	6.5	
SL-P1		6	1 666	9.4	790	6.7	1 600
SL-P1		7	1 666	9.8	793	6.8	
SL-P1		8	1 666	9.7	794	6.6	
SL-P1		9	1 666	9.4	794	6.3	1 700
SL-P1		10	1 666	9.0	794	6	1 700
SL-P1		11	1 666	9.1	780	5.7	
SL-P1		12	1 666	8.3	763	5.2	
HFR-K3	A	1	16 400	7.5	1 200	4	1 600
HFR-K3	B	2	16 400	10	920	5.8	
HFR-K3	B	3	16 400	10.6	920	5.9	1 800
HFR-K3	C	4	16 400	9	1 220	4.9	
HFR-K5	A	1	14 580	7.8	923	4	
HFR-K5	B	2	14 580	10.1	909	5.8	
HFR-K5	B	3	14 580	10.3	903	5.9	
HFR-K5	C	4	14 580	9.3	921	4.9	
HFR-K6	A	1	14 580	8.3	1 090	3.2	
HFR-K6	B	2	14 580	10.6	1 130	4.6	
HFR-K6	B	3	14 580	10.9	1 140	4.8	
HFR-K6	C	4	14 580	9.9	1 130	4.5	
Total Number of Particles			241 376	9.8	1 010	5.3	Averages

Note: * In the German irradiation test program of HTR fuels, position 1 is always the top position. SL-P1 is an exception with position 1 at the bottom.

Table 20 lists the German fuel specimens that were irradiated in the DIDO reactor in Germany. The sphere irradiation rigs (FRJ2-Kxx) were located in the reflector region to accommodate the sphere diameter, resulting in very low fast fluence. The compact irradiations (FRJ2-Pxx) were irradiated in the core region with correspondingly higher fast fluence, but still a softer spectrum than PBMR.

Table 20: Low Fluence Irradiations in Germany (FRJ2)

Experiment	Capsule	Compact or Sphere	Number of Particles	Burnup	Irradiation Temperature	Fast Fluence	Heating Temperature
FRJ2-K13	A	1	16 400	7.5	1 125	0.2	
FRJ2-K13	A	2	16 400	8	1 150	0.2	1 600
FRJ2-K13	B	3	16 400	7.9	1 150	0.2	
FRJ2-K13	B	4	16 400	7.6	1 120	0.2	1 600
FRJ2-K15	A	1	9 560	14.1	970	0.2	
FRJ2-K15	B	2	9 560	15.3	1 150	0.2	
FRJ2-K15	C	3	9 560	14.7	990	0.1	
FRJ2-P27	1	2	2 424	7.2	1 080	1.4	
FRJ2-P27	1	3	2 424	7.6	1 080	1.4	
FRJ2-P27	1	4	2 424	7.6	1 080	1.4	
FRJ2-P27	2	9	2 424	8	1 320	1.7	
FRJ2-P27	2	10	2 424	8	1 320	1.7	
FRJ2-P27	2	11	2 424	8	1 320	1.7	
FRJ2-P27	3	14	2 424	7.6	1 130	1.3	
FRJ2-P27	3	15	2 424	7.6	1 130	1.3	1 400
FRJ2-P27	3	17	2 424	7.2	1 130	1.3	
Total Number of Particles:			116 096	9.4	1 119	0.4	Averages

The burnup values listed for irradiations HFR-P4 and SL-P1 were adjusted to be consistent with the results of the Seibersdorf burnup measurements on selected compacts. Two independent burnup measurement methods produced nearly identical results that were, on average, approximately 12% less than the FZJ gamma scan measurements that were performed on all of the compacts. The Seibersdorf results were used for the three compacts measured, and the other compact burnups were reduced by the average difference. This approach is conservative relative to using the FZJ data, and was chosen because the Seibersdorf measurements are considered by some to be more accurate, although questions remain regarding the limited number of particles used in the Seibersdorf measurements. The MTR irradiations listed above included a total of 357 000 particles with an average burnup, fast fluence, and temperature of 9.4% FIMA, 3.9×10^{21} n/cm² and 1 045 °C.

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