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Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO)-Coated Particle Fuel Performance

Topical Report EPRI-AR-1(NP)-A

2020 TECHNICAL REPORT

Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO)-Coated Particle Fuel Performance

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3002019978

Final Report, November 2020

EPRI Project Managers
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All or a portion of the requirements of the EPRI Nuclear Quality Assurance Program apply to this product.

YES (NO)

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NRC SAFETY EVALUATION

In accordance with U.S. Nuclear Regulatory Commission practice, the NRC safety evaluation immediately follows this page. Other pertinent NRC and EPRI correspondence are included in appendices.

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

August 11, 2020

Craig Stover
Program Manager, Advanced Nuclear
Technology
Electric Power Research Institute Inc
1300 West W.T. Harris Boulevard
Charlotte, NC 28262

SUBJECT: FINAL SAFETY EVALUATION FOR ELECTRIC POWER RESEARCH INSTITUTE TOPICAL REPORT (EPRI), "URANIUM OXYCARBIDE (UCO) TRISTRUCTURAL ISOTROPIC (TRISO) COATED PARTICLE FUEL PERFORMANCE: TOPICAL REPORT EPRI-AR-1(NP)"

Dear Mr. Stover:

By letter dated May 31, 2019, Electric Power Research Institute (EPRI) submitted for U.S. Nuclear Regulatory Commission (NRC) staff review topical report (TR), "Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO) Coated Particle Fuel Performance: Topical Report EPRI-AR-1(NP)" (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19155A173). The NRC staff discussed a preliminary set of questions with EPRI during a public meeting on December 9, 2019 (ADAMS Accession No. ML20029E871) and subsequently issued requests for additional information (RAIs) on January 2, 2020 (ADAMS Accession No. ML20009E065). By letters dated February 26, 2020 and March 9, 2020, EPRI submitted responses to the RAIs (ADAMS Accession Nos. ML20058A040 and ML20071D143, respectively).

The enclosed final safety evaluation (SE) addresses the applicability of TR EPRI-AR-1(NP). The NRC staff has found that the TR is acceptable for referencing in licensing applications to the extent specified under the limitations and conditions delineated in the TR and the enclosed SE.

According to the guidance provided on the NRC website, we request that EPRI publish the accepted version of this TR within three months timeframe after the final SE is issued by NRC. The accepted version shall incorporate this letter and the enclosed final SE after the title page. Also, it must contain historical review information, including NRC RAI questions and your responses. The accepted version shall include a "-A" (designating accepted) following the TR's identification symbol.

As an alternative to including the RAI questions and RAI responses behind the title page, if changes to the TRs were provided to the NRC staff to support the resolution of RAI responses, and the NRC staff reviewed and approved those changes as described in the RAI responses, there are two ways that the accepted version can capture the RAI questions:

1. The RAI questions and RAI responses can be included as an appendix to the accepted version.

2. The RAI questions and RAI responses can be captured in the form of a table (inserted after the final SE) which summarizes the changes as shown in the approved version of the TR. The table should reference the specific RAI questions and RAI responses which resulted in any changes, as shown in the accepted version of the TR.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, EPRI will be expected to revise the TR appropriately or justify its continued applicability for subsequent referencing. Applicants or licensees referencing this TR would be expected to justify its continued applicability or evaluate their plant using the revised TR.

If you have any questions or comments concerning this matter, please contact Jordan Hoellman at (301) 415-5481 or via email Jordan.Hoellman2@nrc.gov.

Sincerely,

/RA/

John P. Segala, Chief
Advanced Reactor Policy Branch
Division of Advanced Reactors and Non-Power
Production and Utilization Facilities
Office of Nuclear Reactor Regulation

Docket No.: 99902021

Enclosure:
Final Safety Evaluation

V

FINAL SAFETY EVALUATION

URANIUM OXYCARBIDE (UCO) TRISTRUCTURAL ISOTROPIC (TRISO) COATED PARTICLE FUEL PERFORMANCE: TOPICAL REPORT EPRI-AR-1(NP)

DOCKET NO. 99902021

1.0 INTRODUCTION

By letter dated May 31, 2019, the Electric Power Research Institute (EPRI), the applicant, submitted for US Nuclear Regulatory Commission (NRC) staff review, "Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO) Coated Particle Fuel Performance, Topical Report EPRI-AR-1(NP)" (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19155A173), hereafter referred to as the topical report (TR).

This TR provides a baseline set of data in order to establish a foundation for TRISO fuel performance, based on testing performed as part of the US Department of Energy (DOE) Advanced Gas Reactor (AGR) Fuel Development and Qualification Program. The TRISO fuel form has applications ranging across a variety of reactor designs. The use of a topical report to establish a well-understood baseline set of fuel performance data for TRISO particles that can be referenced by a variety of vendors presents an efficient means to provide early review in support of potential future applications.

2.0 REGULATORY EVALUATION

This TR does not have a specific regulatory requirement associated with it because how the TRISO fuel meets regulations will depend on how the design and other systems, structures and components (SSCs) are credited in the overall safety of the design. No matter the design, however, 10 CFR 50.34(a)(3)(i) requires, in part, that an applicant for a construction permit to build a power reactor provide principal design criteria (PDC) for the facility. Similar regulatory requirements exist for design certification applications, combined license applications, and standard design approvals (10 CFR 52.47(a)(3)(i), 10 CFR 52.79(a)(4)(i), and 10 CFR 52.137(a)(3)(i), respectively). The PDC establish requirements for SSCs and based on other approaches proposed for advanced reactor designs utilizing TRISO fuel, including the Next Generation Nuclear Plant (NGNP) project, designs with TRISO fuel have used a safety strategy focused on the radionuclide retention capabilities of the TRISO particles.

General Design Criterion (GDC) 10, "Reactor design", in Appendix A of 10 CFR Part 50 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. Although GDC 10 applies only to light water reactor (LWR) designs, the staff expects non-LWR designs to have a similar PDC. Examples of substitute PDC can be found in Regulatory Guide (RG) 1.232 "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors", which provides guidance for developing PDC for non-LWR designs. Establishing fuel design limits and ensuring these limits are not exceeded represent a fundamental underpinning of the safety assessment of a nuclear power plant required by 10 CFR 50.34(a)(1). This TR forms the basis for establishing the design limits for TRISO fuel.

Further, 10 CFR 50.34(a)(1)(ii)(C) requires an applicant to describe the extent to which the reactor incorporates unique, unusual, or enhanced safety features having a significant bearing on the probability or consequences of accidental release of radioactive materials. TRISO fuel presents a unique safety case in using a "functional containment" approach for reducing the release of radioactive materials, and the mechanisms by which TRISO fuel restricts the release of radioactive materials are described in this TR. Such an approach would also likely impact any PDC proposed for containment, but this is outside the scope of this TR.

3.0 TECHNICAL EVALUATION

3.1 INTRODUCTION

Section 1 of the TR summarizes the intended purpose of the report: to establish a baseline set of fuel performance criteria related to TRISO particles, based on the AGR-1 and AGR-2 tests and the irradiation, safety testing, and post-irradiation examination (PIE) results of that testing. The applicant notes this TR could support TRISO development in some of the following areas: to provide for early acceptance and resolution of technical information and data for fuel performance validation; to identify open issues related to the fuel form that could be resolved in subsequent submittals; and to progress fuel performance demonstration in the context of other licensing areas, such as source term and functional containment.

The introduction also states that the applicant requests the staff agree with the following conclusions in the TR (further detailed in the conclusions section and Section 3.8 of this safety evaluation):

- Testing of UCO TRISO-coated fuel particles in AGR-1 and AGR-2 constitutes a performance demonstration of these particle designs over a range of normal operating and off-normal accident conditions. Therefore, the testing provides a foundational basis for use of these particle designs in the fuel elements of TRISO-fueled high-temperature reactor (HTR) designs (that is, designs with pebble or prismatic fuel and helium or salt coolant).
- The kernels and coatings of the UCO TRISO-coated fuel particles tested in AGR-1 and AGR-2 exhibited property variations and were fabricated under different conditions and at different scales, with remarkably similar excellent irradiation and accident safety performance results. The ranges of those variations in key characteristics of the kernels and coatings are reflected in measured particle layer properties provided in Table 5-5 from AGR-1 and AGR-2. UCO TRISO-coated fuel particles that satisfy the parameter envelope defined by these measured particle layer properties in Table 5-5 can be relied on to provide satisfactory performance.
- Aggregate AGR-1 and AGR-2 fission product release data and fuel failure fractions, as summarized in this report, can be used to support licensing of reactors employing UCO TRISO-coated fuel particles that satisfy the parameter envelope defined by measured particle layer properties in Table 5-5 from AGR-1 and AGR-2.

Finally, the applicant notes that "Sections 1–4 and Appendices A and B are included as historical background and context only" and the conclusions of this TR do not rely upon the historical pre-AGR data. This is important in evaluating the context of Sections 2 through 4 of the TR, which are not reviewed for approval as part of this submission.

3.2 REGULATORY BASES

Section 2 of the TR describes how the applicant perceives the TR fits in the broader regulatory framework. The applicant provides background on previous interactions regarding the TRISO fuel form under the NGNP project. Appendix A of the TR details the regulations and guidance that the applicant found to be related to TRISO fuel. The applicant notes that establishing PDC for a reactor is a key part of the licensing basis for a reactor design, but the GDC were developed for LWR designs. Due to this, to assist advanced reactor designers in developing PDC for non-LWR designs, the NRC issued RG 1.232. The applicant states this TR is intended to provide background for meeting Modular High-Temperature Gas-Cooled Reactor Design Criteria (MHTGR-DC 10), "Reactor Design" and MHTGR-DC 16, "Containment Design" for TRISO-fueled designs in Appendix C of RG 1.232.

Specifically, because TRISO fuel is expected to be a primary fission product barrier, it will both be the design feature restricting radioactive releases in accordance with MHTGR-DC 10 and play a large role in the functional containment concept as described in MHTGR-DC 16 and in SECY-18-0096, "Functional Containment Performance Criteria for Non-Light-Water-Reactors." The applicant notes the scope of this TR is foundational, addressing the performance data obtained in the AGR-1 and AGR-2 tests and focused on the TRISO particle. Further fuel qualification efforts for other aspects of the fuel or a broader scope of performance conditions could be within the scope of future submittals, as needed.

3.3 TRISO-COATED PARTICLE FUEL EXPERIENCE BASE

Section 3 of the TR provides a review of historical experience related to the development of TRISO particles, including in the UK, US, Germany, and Japan. Further experience from these nations and others is described in Appendix B of the TR. The data in Section 3 of the TR is presented for context and background, only to show that TRISO fuel meeting prismatic HTR fuel performance requirements can be fabricated, but this information is not used by the NRC staff as part of the evaluation in this document.

3.4 FISSION PRODUCT RETENTION, PARTICLE DESIGN, AND PERFORMANCE BASES

Section 4 of the TR describes the features of the TRISO fuel design that make up the design and performance envelope. In the TR, the applicant explains how the concept of functional containment applies to the TRISO fuel design. Functional containment is discussed both in RG 1.232 and SECY-18-0096, and the applicant uses the definition in RG 1.232: "a barrier, or set of barriers taken together, that effectively limit the physical transport and release of radionuclides to the environment across a full range of normal operating conditions, anticipated operational occurrences, and accident conditions."

Generally, a collection of barriers is relied upon to ensure offsite dose limits are not exceeded. For high-temperature reactors using TRISO fuel, the fuel particle will be the primary barrier credited, and so fuel performance established for the TRISO particles must demonstrate a low fuel defect and failure frequency. This is accomplished in large part through the design of the particles themselves – a series of coatings surrounding each of the individual fuel kernels act together to retain most fission products and transfer heat effectively, while the particles are compatible with a carbonaceous matrix that provides a structural form to contain a large number of the small particles. The TRISO particle layers include (sequentially from inner- to outer-most) an inner fuel kernel, a low-density pyrolytic carbon (PyC) buffer layer designed to accommodate

gaseous fission products, and a pair of higher-density PyC (inner pyrolytic carbon (IPyC) and outer pyrolytic carbon (OPyC)) layers that sandwich a silicon carbide (SiC) layer. The IPyC and OPyC act as "load-bearing" components of the particle's effective pressure boundary, and are structurally supported by the SiC layer, which acts as the primary metallic fission product barrier. The SiC layer is compressed by the two higher-density PyC layers, which assists in maintaining the structural integrity of the SiC layer. Different designs use different values for the layer parameters in particle design; these are summarized in Table 4-2 of the TR.

Section 4.3 of the TR explores the different potential failure mechanisms for the TRISO particle, a visual representation of how these failures may occur is shown in Figure 4-4 of the TR. Each of the failure mechanisms identified can be controlled through a combination of manufactured particle properties (e.g. density and layer thicknesses) and reactor service conditions (e.g. temperature, burnup, fluence). One failure mechanism, discussed in more detail later, that can result in releases through even "intact" particles is diffusion-stimulated releases through intact layers, which is a function of time at temperature and burnup, among other parameters. The TR also notes that while as-manufactured heavy metal contamination is not an in-service failure mechanism, it may impact fission product releases.

NRC Staff Evaluation

Staff recognizes that the applicant does not request staff approval of Section 4 of the TR, addressing different potential failure mechanisms, and does not present any conclusions in Section 8 of the TR based solely on this section. However, information related to the fuel particle and performance envelope play an important role in defining the relation between the tested fuel and the requests in this TR. Conclusion 1 of the TR states that "testing of UCO TRISO-coated fuel particles in AGR-1 and AGR-2 constitutes a performance demonstration of these particle designs over a range of normal operating and off-normal accident conditions." Discussions under the conclusion reference a compact-averaged burnup of 7.3-19.6% fissions per initial metal atom (FIMA) and time-averaged maximum temperatures of 1069-1360°C. Other relevant performance parameters that bound the data set, such as those referenced in Figure 4-6 of the TR (packing fraction, fluence, power density) could influence particle performance. In reviewing the TR and during audit discussions (ADAMS Accession No. ML19310F085) with Idaho National Laboratory (INL) experts, staff determined other key parameters include fluence, time-averaged power, kernel stoichiometry, kernel-to-buffer ratio, and particle microstructure. In a letter dated February 26, 2020, the applicant provided clarifications regarding the relevant additional parameters associated with the particle performance These topics are discussed in subsequent sections of this evaluation as limits for TRISO particles. The conclusions of this TR are subject to these limits, supplemented by the limitations and conditions section of this TR.

Staff notes that the above discussion relates to particle performance only, not any effect related to the compact. Compact performance could be credited as an additional fission product retention mechanism, or potentially require additional considerations due to key parameters related to the final fuel form such as particle packing fraction or material properties. Accordingly, staff-imposed Limitation 1 as stated in the Limitations and Conditions section of this evaluation (Section 4.0), related to the scope of this TR covering the TRISO particle only.

In 2005, the DOE established the NGNP project at INL to support near-term commercial deployment of a high-temperature gas-cooled reactor technology demonstration plant. The design and licensing strategy of the NGNP centered on radionuclide retention capabilities of TRISO particle fuel, which included the INL AGR Fuel Development and Qualification Program.

The EPRI TR covers foundational fuel performance testing from the AGR-1 and AGR-2 tests including the irradiation, safety testing, and PIE results. The NRC staff assessment of "Next Generation Nuclear Plant Quality Assurance Program Description [QAPD]," dated September 12, 2012 (ADAMS Accession No. ML12241A157), found that the QAPD was acceptable for use during the technology development and high-level design phase of the NGNP project. Because the TR did not describe the scope of quality assurance (QA) activities performed by INL to obtain and submit the data used in the EPRI TR, the staff sought additional clarification in draft request for additional information (RAI) 5. In a letter response dated March 9, 2020 (ADAMS Accession No. ML20071D143), EPRI stated that these research and development activities are associated with "technology development" activities, and that the QA standards reflected in the NGNP QAPD and assessed by the NRC staff were implemented during the performance of those activities. The staff reviewed EPRI's response and concludes that the activities involved in developing the data referenced in this report performed by INL are bound by the NRC approved QAPD in that the activities were associated with technology development and high-level design activities.

3.5 ADVANCED GAS REACTOR FUEL DEVELOPMENT AND QUALIFICATION PROGRAM

Section 5 of the TR summarizes the AGR program and its relation to the TRISO particle design envelope discussed in the TR. Design specifications for the fuel and operational campaign were informed by US and international experience and expected industry performance requirements, and the AGR program was intended to provide for fuel qualification data in support of HTR designs As a whole, the program focused on the following topical areas related to fuel qualification: fuel fabrication, fuel and material irradiation, fuel PIE and safety testing, fuel performance modeling, fission product transport, and source term development

The AGR program consists of a series of campaigns, which cover a range of conditions from initial fuel scoping, to fuel performance, to fission product transport, to safety performance and accident condition testing The TR provides a brief discussion related to TRISO fuel fabrication methods used to produce the particles for the AGR-1 and AGR-2 tests; the TR emphasizes that the conclusions outlined in the TR are intended to be fabrication method independent, and that only the fuel specifications (provided in Table 5-5 of the TR) constrain the performance as demonstrated in the AGR tests. Data related to the particles produced for these tests can be found in Tables 5-1, 5-2 and 5-3 of the TR.

The TR provides a brief discussion related to particle characterization, including methodology and particle makeup Values obtained from whatever set of characterization methods (for the AGR-1 and AGR-2 tests) is used then fall within the particle property bounds in Table 5-5. The TR notes that the final step in fuel preparation is fabrication of the final fuel form – in the case of the AGR program, a cylindrical compact The final fuel form is generally composed of a graphite binder due to satisfactory material properties and compatibility with the outer TRISO layer.

The TR also provides a supplementary discussion related to particle carbon content of the UCO particles. The particles used in the AGR testing targeted a uranium carbide content of approximately 30% to achieve burnups of 20% FIMA without exceeding acceptable CO gas formation. The TR provides an extended discussion from the literature on the relative effects of carbon content on gas formation

NRC Staff Evaluation

The primary conclusions included as part of the TR center around Table 5-5, which is implied to provide an exclusive set of properties that, if determined to be satisfied, ensure that the manufactured TRISO fuel will perform to the same standards as the particles tested in AGR-1 and AGR-2. Based on the AGR-1 and AGR-2 tests and the consequential information provided in the TR, particles behaved similarly even with different manufacturing methods. International experience with a broader set of manufacturing methods lends further credence to the assertions in Section 5.3.6 of the TR, which states, "there is not a unique set of kernel specifications that are critical to successful TRISO fuel." Based on the information provided in the TR, staff agrees that it is reasonable and possible to establish a set of measurable performance criteria independent of the manufacturing process based on the AGR experience and informed by historic TRISO development.

Effectively, the TR ties the property specification envelope to the performance demonstration of the AGR-1 and AGR-2 tests as discussed in Sections 6 and 7 of the TR. However, it was not clear in the TR that the exclusive set of parameters referenced in Table 5-5 was sufficient to demonstrate that manufactured fuel was sufficiently similar to the tested AGR fuel. Table 5-5 of the TR listed only kernel layer properties and thicknesses, along with properties related to relative uniformity of the particle (IPyC and OPyC anisotropy and particle aspect ratio). Other elements that the TR highlights as important, but that are not directly referred to in Table 5-5, include kernel-to-buffer ratio for the fuel particle (and potentially its associated size), columnar grain structure of the SiC, and carbon content of the UCO.

In the letter response dated February 26, 2020, the applicant added a number of additional areas of discussion related to key aspects of the tested TRISO particles. Section 5.3.2.4 of the TR provides additional context related to the SiC microstructure. Although the AGR program did not include quantitative limits on grain size, testing experience has shown that grains that are sufficiently large or columnar in nature (effectively those that provide for less torturous pathways for fission product escape) could fail to perform as intended. Because it would be challenging to establish a limit value, no restriction has been included in Table 5-5, but the TR provides a visual example of what constitutes approximate upper bound on an acceptable grain size in Figure 5-2, and the expectation is that an applicant referencing this TR would institute a similar control on manufactured TRISO particles.

Section 4.2 and Section 4.3 of the TR highlight the importance of SiC stresses in the mechanical fuel performance of the TRISO particles. Stresses are captured in a stress metric (σ in the TR, used as a proxy for tensile stress in the SiC layer), which encapsulates kernel stresses as a function of kernel and buffer volume, as well as burnup, SiC radius, and SiC thickness. Using the AGR data and Monte Carlo simulation techniques, the applicant calculated the value for σ and examined the distribution of stresses for the tested particles. This distribution is provided in Table 5-6 of the TR and is similar to the values used for historical fuel tests. Because σ includes considerations related to different particle sizes and layer parameters, it is a reasonable value for demonstrating the mechanical efficacy of potentially dissimilar particles (similar to non-dimensional values used in scaling analysis).

The values in Table 5-6 of the TR are not indicated as limits on the applicability of the TR. Although most of the parameters in σ are effectively derived from other limits cited in the TR (e.g. Table 5-5 plus burnup), kernel size itself is not. Staff recognizes this TR demonstrates TRISO fuel particle performance over a range of fabrication and operation conditions captured within the AGR-1 and AGR-2 data. The staff expects that Table 5-5 of the TR adequately

captures the coating properties that bound acceptable particle performance based on the provided data. However, staff notes that fuel kernel size can differ among different designs (see Table 4-2 of the TR), and this is not captured in the TR. Accordingly, the staff-imposed **Condition 1** (discussed in Section 4 0 of this evaluation) on the TR, related to ensuring the AGR particle sizes sufficiently envelope those used by applicants or licensees referencing this TR. Staff notes this discussion in the TR also captures the considerations related to kernel-to-buffer ratio discussed earlier in the TR.

The TR states that "fuel particles tested in AGR-1 and AGR-2 exhibited property variations. .with remarkably similar excellent irradiation and accident safety performance results The ranges of those variations in key characteristics of the kernels and coatings are reflected in measured particle layer properties provided in Table 5-5 from AGR-1 and AGR-2." Table 5-5 provides a set of characteristics for "acceptable" TRISO particles, accounting for the bulk of the particle sample and the tolerance extremes for particles at the tails of the distribution. Staff audited (ADAMS Accession No ML19310F085) the data and documentation supporting the values found in the TR. The staff issued RAI 3 on January 2, 2020, requesting the applicant provide, in part, a table with a clear requested range for each property for approval to be referenced in the conclusions. The staff reviewed the RAI response, dated February 26, 2020, and found the values in the revised Table 5-5 of the TR accurately reflect the data gathered. Based on the provided data, there is a clear basis for use of the measured values in Table 5-5.

In the response to RAI 3 (ADAMS Accession No. ML20058A040), the applicant added Appendix C of the TR to provide further context for the fuel manufacturing specifications used for the AGR program The specification range is larger than the tested fuel range (from Table 5-5), sometimes substantially This is an important consideration and a notable distinction of TRISO fuel as opposed to traditional previously licensed fuel designs – in TRISO fuel, there are many orders of magnitude more individual fuel elements, a very small fraction of which will be statistically expected to fail. The net effect of this small number of expected failures, as part of a broader functional containment approach for the full design (as described in SECY-18-0096), allows for a more granular performance-based approach (because these "anticipated" failures can be quantified directly and accounted for) with potentially larger margins of safety. Because this design philosophy for TRISO fuel differs dramatically from existing practice, staff recognizes and agrees with the statement in the TR that:

> "The values in Table 5-5 are not intended to define a comprehensive envelope of TRISO fuel that is expected to have acceptable performance The data characterize the range of properties for particles that performed well during the AGR-1 and AGR-2 irradiations, but do not define the only ranges or combination of ranges that would perform well under these irradiation conditions or under service conditions proposed by fuel fabricators and reactor designers."

Particles from the AGR tests fall within the ranges delineated by the statistical ranges laid out in the TR (specifically, the specification ranges reflected in Table 5-5). As such, no limitation or condition need be imposed on an applicant directly referencing this TR. However, staff notes that this TR does not provide for the only set of parameters for acceptable TRISO performance, and that TRISO particles sharing many but not all of the characteristics of the particles tested in the AGR program could easily be shown to perform adequately. An applicant could have particles exceeding boundary values in areas and continue to have acceptable performance, but an applicant-specific demonstration would need to be provided. Staff is receptive to reviewing applications that reference this TR with supplemental justification for limited discrepancies from

the performance envelope described here, and the review of that supplemental justification would be incorporated as part of any future licensing submittal.

As stated previously, the conclusions in this TR are limited to the TRISO particle Performance of the final fuel form, including fission product retention and any other functional performance of the fuel itself is not within the scope of this TR and would need to be the subject of a future submittal.

As presented in the TR and supplemented by the response to RAI 2, the staff agrees that the tested carbon content during the AGR program does not represent a lower (or upper) bound on the amount of carbon that could be used in an acceptable TRISO particle. There is a lower bound, based on testing and the literature presented, and the precise value of that bound is based on burnup (and potentially temperature at elevated temperatures) and cannot be clearly established. Staff agrees with the assertion in the TR that there is a "wide range of UO_2/UC_2 ratios that maintain effectiveness at (a) limiting CO gas formation and (b) promoting the formation of rare earth oxides over the formation of rare earth carbides in order to increase retention of rare earths in the kernel." Due to the difficulty in establishing a clear boundary value, staff expects an applicant referencing this TR to provide a target burnup and carbon content range within the boundaries provided for in Figure 5-4 of the TR. This is reflected in Limitation 2 as stated in the Limitations and Conditions section of this evaluation (Section 4.0).

3.6 AGR-1 AND AGR-2 IRRADIATIONS

Section 6 of the TR provides an extended discussion of the AGR-1 and AGR-2 irradiation programs The TR outlines the experimental setup, layout including instrumentation and gas lines for cooling and fission product monitoring, and calculated power profiles over the course of the irradiation. Figure 6-6 and Figure 6-7 of the TR show the calculated capsule-average heat generation rate in the experimental compacts versus time. Many of the capsules showed an increase in power during the first half of the experiment; the applicant explains this is due to the depletion of the boron burnable poison added to the graphite fuel holders. The applicant states "capsule-average burnups [for AGR-1] ranged from 13.4% FIMA in Capsule 6 to 18.6% FIMA in Capsule 3" and "for AGR-2... capsule-average burnups ranged from 9.3% FIMA in Capsule 6 to 12.2% FIMA in Capsule 2 for UCO".

Further data is provided for power density, burnup, fluence, temperature, and other parameters for both the AGR-1 and AGR-2 campaigns. Direct fuel performance can be assessed through use of fission product release over birth (R/B) ratios measuring a collection of krypton and xenon isotopes. Calculated R/B ratios showed an "extremely low" gas release for AGR-1 and indicated zero fuel failures experienced out of 300,000 particles. AGR-2 had slightly higher R/B ratios (partially due to higher uranium contamination in the compacts), such that some small number of particle defects or failures cannot be precluded. Up to four failures were present out of 114,000 UCO particles in the AGR-2 tests, a failure fraction still below the design specification. Based on the results presented, the TR concludes that the AGR-1 and AGR-2 tests demonstrate excellent performance of UCO TRISO-coated particles with significant margin. These particles were fabricated using different conditions and properties (confined to the values provided in Section 5 of the TR).

NRC Staff Evaluation

Staff agrees that TRISO particles were tested over a range of fluxes, temperatures, and burnup values during the AGR-1 and AGR-2 testing. Based on the above discussion related to

Section 5 of the TR, the data collected is applicable to the particles manufactured to the specifications in Table 5-5 of the TR. Data presented in the TR is difficult to summarize with a single bounding value due to the nature of the fuel form and experimental setup (many individual particles located throughout a series of different test environments), but the temperature data presented in Figures 6-26 through 6-28 of the TR provide a reasonable effective temperature profile to reference. Further, power density over time for the tests is presented in Figures 6-6 and 6-7 of the TR, and burnup and fluence are captured in Figures 6-9 through 6-11 of the TR. Ultimately, the performance boundaries for the scope of this TR are provided in Table 6-6 and Figure 6-30. The discussion related to Conclusion 1 in the TR accurately reflects these bounds. These data provide a "performance envelope" for the collection of AGR-1 and AGR-2 tests.

Staff reviewed the values provided as part of the TR and audited (ML19310F085) the underlying test results and methods for examining the test specimens and found them to be without issue. Staff compared the parameters used to historical fuel performance parameter thresholds for both traditional reactor fuel forms and for previous TRISO fuel experiments. Due to the nature of the TRISO particles and the substantial differences in form from most previously licensed fuel designs, staff agreed that the parameters chosen (burnup, fluence, and time-averaged temperature, power density, and particle power) represent an adequate set for evaluating TRISO fuel performance. This does not represent a complete datum for evaluating fuel performance in all operational modes – as noted earlier, performance of the final fuel form remains the responsibility of a future license applicant, and staff expects some transient and accident conditions may exhibit behavior that yields values that exceed the envelope presented in the TR. Nevertheless, staff agrees that the testing provides a foundational basis (and a valid data set) for use in future licensing submittals using TRISO particles.

As stated previously, staff found that the activities performed by INL are bound by the NRC approved NGNP QAPD in that the activities were associated with technology development and high-level design activities. As such, staff finds that the data referenced in the TR can be used to inform fuel performance for TRISO particles for those referencing this TR, subject to the parameter envelope in Table 5-5 of the TR.

NRC staff notes that fission product release measurements are limited to fission gas release during irradiation (discussed in Section 6.7 of the TR) and long-lived post-irradiation fission product release data (discussed throughout Section 7 of the TR). No data is obtained on short-lived radioisotopes during post-irradiation examination, because decay reduces their inventory to undetectable levels by the time PIE is initiated. Fuel performance criteria related to short-lived non-gaseous fission products are not captured by the test program described in the TR and are therefore not part of the scope of the TR. Accordingly, staff-imposed Condition 3 (discussed in Section 4.0 of this evaluation), and an applicant referencing this TR will need to consider the impact, if any, of non-gaseous short-lived fission products.

3.7 ASSESSMENT OF FUEL PERFORMANCE FROM POST-IRRADIATION EXAMINATION AND SAFETY TESTING

Section 7 of the TR characterizes the data obtained from the PIE following the AGR-1 and AGR-2 tests as well as the safety testing performed where the irradiated compacts were heated to higher temperatures. The TR summarizes the different methods that are used to quantify fission product releases (by either quantifying the fission product inventory remaining in the fuel, or the inventory that has been released from the fuel)

The primary longer-lived fission products that are of concern are Silver-110 m, Cesium-134, Europium-154 and -155, and Strontium-90. Cerium-144 and palladium are also discussed in the TR but were generally not indicative of large amounts of release. Results show that cesium was released from some particles in the event of SiC failures, and much better retained for compacts where no SiC failures were observed. Europium and strontium were generally detected to a higher degree (more than an order of magnitude in terms of fractional release) in compacts as compared to capsules, indicating these isotopes release from the particle as a function of time at temperature but are largely retained in the compact material (dependent on irradiation and temperature conditions). Silver showed higher releases to the capsule, and "at the individual particle level, Ag [silver] release could range from complete retention to complete release."

Evaluation of the individual irradiated particles indicate most particles experienced debonding of the buffer and IPyC layer, and some particles (roughly a quarter in AGR-1) experienced fracture of the buffer layer. However, failure of the SiC layer (the primary TRISO barrier) occurred very rarely – approximately one per 52,000 particles in the irradiation testing and one per 15,000 particles at 1600°C testing following irradiation. Failure of the TRISO particle itself was even less common than the SiC failure. Examination of the particles for SiC failure mechanisms showed one primary mechanism for failure: buffer shrinkage leading IPyC fracture. The TR states this failure mechanism is likely to be difficult to model.

Testing at elevated temperatures following irradiation was conducted (referred to in the TR as "safety testing"); the results from this testing are shown in Figures 7-6 through 7-8 of the TR. Testing was conducted at hundreds of hours at temperatures ranging from 1600°C to 1800°C. Results related to cesium showed that cesium was unlikely to be released except in the case of SiC failure, where particles could release a relatively large fraction of contained cesium. Nearly all safety tests showed a large initial release of silver as the compact was heated, followed by a leveling off of the silver release; this was attributed to the "depletion" of the silver inventory in the compact outside of intact SiC layers. Europium and strontium behaved similarly, and generally releases of these two elements increased with increases in temperature and time at temperature.

A statistical summary of SiC and total TRISO failure is provided in Table 7-2 and Figure 7-15 of the TR. The TR reports the AGR-1 failure fraction as $\leq 1.1 \times 10^{-5}$ at 95% confidence and a conservative failure fraction for AGR-2 as $\leq 8.1 \times 10^{-6}$. Failure fractions increased at increasing temperatures beyond 1600°C in the safety testing. SiC failures were still rare, though more common. the total (95% confidence upper bound) SiC failure fractions across the AGR-1 and AGR-2 tests were $\leq 3.6 \times 10^{-5}$ during irradiation and $\leq 1.7 \times 10^{-4}$ and $\leq 1.3 \times 10^{-3}$ during safety testing at 1600°C and 1800°C, respectively. The applicant states these data provide for a performance demonstration across a range of conditions, and can be used to support the licensing of UCO TRISO-fueled reactors, subject to the parameters discussed in Section 5 of the TR.

NRC Staff Evaluation

NRC staff agrees with the above summary. The testing referenced above was performed under an adequate QA program, and the data gathered is sufficient to draw the conclusions that are within the scope of this TR. As such, NRC staff finds the collected data is valid for use in qualifying TRISO particles subject to the property specification and performance envelope outlined in the TR. Failure data for irradiated TRISO particles meeting the envelope described can be used directly in referencing this TR.

"Safety testing" values for testing at elevated temperatures, however, should not be used directly. Staff agrees the data is valid, but it may or may not directly apply to the specific scenarios calculated for any given design A conservative assessment of the statistical failure data presented in this TR at lower temperatures (roughly less than 1600°C) may be directly used empirically. Once particles reach the temperature condition that shows failures may occur (that is, roughly greater than 1600°C), an applicant or licensee referencing this TR would need to justify how transient accident conditions were bounded by the data provided in the TR or provide additional analyses or testing to justify fuel performance in the specific accident scenario. Further, transient conditions (like sharp power ramp rates) are not generally within scope of this TR and would again require justification.

Staff further notes that this TR provides for an empirical, experimental data set pertaining to particle failure in a limited temperature and power regime. The data do not enable a deterministic prediction of individual particle failure; instead, based on an assessment of the fuel operating conditions (e.g., temperature) outlined in the TR, a statistical failure probability for a population of particles can be projected. Thus, the TR provides a reasonable data set to establish a performance envelope for particles that will not be expected to fail at rates exceeding the statistically calculated values. The results discussed in Section 7.4 of this TR provide for an understanding of TRISO particle failure mechanisms. The dominant SiC failure mechanism in the AGR-1 and AGR-2 tests is different from the failure mechanisms observed in past TRISO fuel irradiations and from mechanisms currently embedded in fuel performance models. The staff emphasizes that this TR does not provide sufficient data to evaluate simulated individual particles directly. The TR, can, however, be used to empirically evaluate failure probabilities of general populations of TRISO particles manufactured to the specifications outlined in Table 5-5 and subject to the performance envelope defined by Table 6-6 and Figure 6-30 of the TR. Accordingly, staff imposed Condition 2 (discussed in Section 4.0 of this evaluation) related to the use of particle failure data based on the information presented in this TR.

With respect to the quantitative effects of the SiC microstructure discussed in Section 5.3 of the TR and in the above evaluation, the applicant found no pronounced differences in performance up to 1700°C due to differing SiC microstructures subject to the visual controls discussed earlier. Based on the data, staff agrees with that assessment and it supports the conclusions made above regarding the applicability of the AGR data.

3.8 SUMMARY/CONCLUSIONS

The applicant requests the NRC approve the following conclusions, based on the information presented in Sections 5 through 7 of the TR:

- Testing of UCO TRISO-coated fuel particles in AGR-1 and AGR-2 constitutes a performance demonstration of these particle designs over a range of normal operating and off-normal accident conditions. Therefore, the testing provides a foundational basis for use of these particle designs in the fuel elements of TRISO-fueled HTR designs (that is, designs with pebble or prismatic fuel and helium or salt coolant).
- The kernels and coatings of the UCO TRISO-coated fuel particles tested in AGR-1 and AGR-2 exhibited property variations and were fabricated under different conditions and at different scales, with remarkably similar excellent irradiation and accident safety performance results. The ranges of those variations in key characteristics of the kernels and coatings are reflected in measured particle layer properties provided in Table 5-5 from AGR-1 and AGR-2. UCO TRISO-coated fuel particles that satisfy the parameter

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

NRC Staff Evaluation

Conclusion 1 of the TR requests acceptance that the AGR-1 and AGR-2 tests constitute a performance demonstration of AGR TRISO particles such that the testing forms a foundational basis for use in future TRISO fuel designs. The associated discussion under Conclusion 1 provides the performance ranges of the subject particles in terms of burnup, time-averaged temperatures, fast neutron fluence, and power density. Coupled with the time-averaged particle power – important because it accounts for differences in compacts and thus focuses on the particles themselves – discussed in Section 6 of the TR, staff agrees this set of performance parameters adequately captures the envelope of the AGR-1 and AGR-2 test conditions, as discussed above. Subject to the conditions laid out in the limitations and conditions discussed in this evaluation, staff finds Conclusion 1 to be applicable and acceptable.

Conclusion 2 of the TR requests broader approval: particles manufactured consistent with the limits identified in Table 5-5 of the TR, which are based on the calculated AGR-1 and AGR-2 particle parameters, can be relied on to provide satisfactory performance. Satisfactory performance, in this case, is aligned with the empirical performance demonstration described in Sections 6 and 7 of the TR. The discussion supporting Conclusion 2 further highlights the differences in coating conditions and manufacturing methods, with the exception that coating in all cases was carried out using an uninterrupted process to ensure high coating quality. As discussed above, the applicant provided a clear basis for use of the values in Table 5-5, and the values in Table 5-5 are not representative of an exclusive set of parameters for acceptable TRISO performance. However, the values in Table 5-5 serve to tie the empirical data discussed in this TR to the tested AGR-1 and AGR-2 particles, and so the staff finds the scope of Conclusion 2 acceptable. Subject to the limitations and conditions discussed in this evaluation, staff finds Conclusion 2 to be applicable and acceptable.

Conclusion 3 requests the ability to use the AGR-1 and AGR-2 fuel failure and fission product release data to support licensing TRISO-fueled reactor designs that satisfy the parameter envelope defined in Table 5-5 of the TR. The supporting discussion notes this conclusion is limited to the isotopes discussed in Section 6.7, 6.8, 7.1 and 7.3 of the TR – that is, short-lived fission gases and longer-lived isotopes (Cs, Eu, Sr, Ag, Kr) discussed in greater detail in Section 7 of the TR. Based on the provided data, 95% confidence interval failure fractions for the irradiation testing are provided, and staff finds these values can be used by applicants referencing this TR. Relative values for radionuclide releases are confined to intact particles, and demonstration of any retention within the fuel form outside the particle is the responsibility of the applicant or licensee referencing this TR, as stated in Limitation 1 of this evaluation.

As discussed in the above evaluation, safety testing data post-irradiation should only be used considering the context of the specific design and the applicability of the data to the expected fuel conditions. The data itself is valid, but transient and accident conditions may or may not match the conditions experienced by the tested AGR particles. Thus, justifying the applicability of the safety testing data is an exercise left to an applicant or licensee referencing this TR.

4.0 LIMITATIONS AND CONDITIONS

The staff imposes the following limitations and conditions with regard to the TR and its conclusions:

(1) Limitation 1: The scope of this TR applies only to the UCO TRISO particles themselves. How the final fuel form is qualified and any impacts of the fuel form or other influences of the specific reactor design beyond the fuel form on the holistic fuel performance (for instance, any uranium contamination in the compact material) is the responsibility of the vendor or designer referencing this TR.
(2) Limitation 2: This TR applies only to UCO TRISO particles that fall within the ranges discussed in Section 5.3 of the TR. If an applicant chooses to use UO_2/UC_2 ratios or burnup values that differ meaningfully from those used in the AGR program, the applicant must provide a justification for how the burnup and carbon content ratios conform to the performance ranges discussed in Section 5.3 of the TR.
(3) Condition 1: An applicant or licensee referencing this TR must evaluate any discrepancies between their fuel particles and the TRISO particles used in the AGR program - specifically, reviewing the ranges specified in Table 5-6 for stress values to capture any effects from different kernel sizes to ensure the data in the TR remain applicable.
(4) Condition 2: The performance limits in Table 6-6 and Figure 6-30 of the TR are the result of different tests with distinct samples, not all of which had the maximum bounds occur during the same test. The test results include considerations for uncertainty discussed in Sections 6.4 and 6.5 of the TR. Further, when failures may occur, the data supporting the TR provides empirical evidence of failure based on aggregate test conditions rather than evidence of individual particle failure. Applicants referencing this TR must ensure that they either remain within the tested bounds or justify how their proposed operating conditions remain applicable, considering uncertainty in both the AGR test results as described in the TR and any analytical uncertainty resulting from the proposed analytical method.
(5) Condition 3: Data discussed in this TR does not consider the impacts of short-lived fission products beyond those captured in the gas phase during experiments. Any applicant or licensee referencing this TR must disposition the impacts, if any, of short-lived fission products on the safety analyses and operational dose considerations, or any other regulatory considerations resulting from short-lived fission products, in addition to the data discussed in the TR.

5.0 CONCLUSION

Based on the evaluation above, staff has concluded that the applicant has demonstrated that TRISO particles produced to the specifications and limited to the performance parameters documented in the TR can be used to satisfy a portion of the requirements associated with GDC 10 or an equivalent PDC, subject to the Limitations and Conditions in Section 4.0 of this evaluation. More specifically, TRISO particles produced to the specifications within the TR (including Table 5-5) and limited to the performance parameters in the TR (including Table 6-6) will perform in accordance with the AGR data presented in Sections 6 and 7 of the TR. This data can be used (subject to the performance thresholds of the AGR tests discussed in the TR) to support safety analyses referencing the unique design features of TRISO particle fuel. Staff

notes, as discussed in this evaluation, that performance of the particle represents only part of the justification needed to support qualification of TRISO-fueled designs – performance characteristics of the final fuel form and how any given design copes with transient scenarios outside the scope of the data presented in the TR will be needed to support any future licensing submittal referencing this TR.

6.0 REFERENCES

1. Transmittal of "Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO) Coated Particle Fuel Performance: Topical Report EPRI-AR-1(NP)" dated May 31, 2019 (ADAMS Accession No. ML19155A173).

2. Audit Report for the Regulatory Audit of EPRI Topical Report, Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO) Coated Particle Fuel Performance, dated November 19, 2019 (ADAMS Accession No. ML19310F085).

3. Request for Additional Information (RAI) Transmittal for Topical Report EPRI-AR-1(NP), Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO) Coated Particle Fuel Performance, dated January 2, 2020 (ADAMS Accession No. ML20009E065).

4. 12/09/2019 Summary of Public Meeting with EPRI regarding Topical Report EPRI-AR-1(NP), Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO) Coated Particle Fuel Performance, dated February 10, 2020 (ADAMS Accession No. ML20029E871).

5. EPRI – Responses to Requests for Additional Information (RAIs) on Topical Report EPRI-AR-1(NP), "Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO) Coated Particle Fuel Performance," dated February 26, 2020 (ADAMS Accession No. ML20058A040).

6. EPRI – Response to Requests for Additional Information (RAI) on Topical Report EPRI-AR-1(NP), "Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO) Coated Particle Fuel Performance," dated March 9, 2020 (ADAMS Accession No. ML20071D143).

7. EPRI Comments on NRC Staff Draft Safety Evaluation for EPRI TRISO Topical Report (ADAMS Accession No. ML20112F498).

8. EPRI Summary of Extent of Condition Changes to TRISO Topical Report (ADAMS Accession No. ML20114E116).

ACKNOWLEDGMENTS

The following organization, under contract to the Electric Power Research Institute (EPRI), prepared this report:

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Development and preparation of this report was sponsored by EPRI and funded in part by the U.S. Department of Energy, Office of Nuclear Energy, through U.S. Government contract no. DE-AC07-05ID14517.

Under this collaborative project, technical content presented in Sections 2 – 7 and Appendices A and B was compiled and prepared specifically for this report with the assistance of Idaho National Laboratory (INL), Battelle Energy Alliance, LLC (BEA), under contract DE-AC07-05ID14517 with the U.S. Department of Energy. This content is documented in INL/LTD-18-46060 Rev. 0, *Technical Bases for the Performance Demonstration of TRISO-coated UCO Fuel Particles* and has been derived from other INL/BEA reports and results. The U.S. government retains a non-exclusive, paid-up, irrevocable worldwide license to publish or reproduce the published form of INL/LTD-18-46060, or allow others to do so, for U.S. government purposes.

EPRI appreciates and acknowledges the substantial support from and contributions by INL technical subject matter experts: P. Demkowicz, M. Holbrook, J. Kinsey, D. Marshall, W. Moe, H. Gouger, and D. Petti.

EPRI also appreciates and acknowledges the support of the following organizations and individuals, working collaboratively under the industry-led High Temperature Reactor Technology Working Group, to support the planning, development, and review of this report:

- Framatome (F. Shahrokhi)
- Kairos Power (P. Hastings, J. Tomkins, B. Collin, D. Peebles, D. Gardner, and M. Hackett)
- StarCore Nuclear (L. Eskin)
- X-energy (G. Bell, P. Pappano)
- Southern Company (C. Medlock)

This report benefitted greatly from technical reviews and comments provided by John Beale (EPRI) and independent reviewers Richard Hobbins and Finis Southworth.

This publication is a corporate document that should be cited in the literature in the following manner:

Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO)-Coated Particle Fuel Performance: Topical Report EPRI-AR-1(NP)-A. EPRI, Palo Alto, CA: 2020. 3002019978.

ABSTRACT

Nuclear fuel, fuel forms, and operating conditions vary widely across the numerous advanced reactor designs under development. However, tristructural isotropic (TRISO)-coated particle fuel is foundational for many high-temperature reactor (HTR) designs, including high-temperature gas-cooled reactors (HTGRs) and fluoride salt-cooled high-temperature reactors (FHRs). The U.S. Department of Energy initiated the Advanced Gas Reactor Fuel Development and Qualification (AGR) Program in 2002 to establish U.S. capability to fabricate high-quality uranium oxycarbide (UCO) TRISO fuel and demonstrate its performance. Results from the first two fuel irradiation tests in the program, designated AGR-1 and AGR-2, demonstrate UCO fuel performance during irradiation and in post-irradiation high-temperature accident safety tests. This report consolidates the technical bases for the functional performance of UCO-based TRISO-coated particles so these particles can be used by a variety of HTR developers in their designs. The U.S. Nuclear Regulatory Commission (NRC) issued a safety evaluation in August 2020 following review and approval of this topical report on the basis of three key conclusions:

1. Testing of UCO TRISO-coated fuel particles in AGR-1 and AGR-2 constitutes a performance demonstration of these particle designs over a range of normal operating and off-normal accident conditions. Therefore, the testing provides a foundational basis for use of these particle designs in the fuel elements of TRISO-fueled HTR designs (that is, designs with pebble or prismatic fuel and helium or salt coolant).
2. The kernels and coatings of the UCO TRISO-coated fuel particles tested in AGR-1 and AGR-2 exhibited property variations and were fabricated under different conditions and at different scales, with remarkably similar excellent irradiation and accident safety performance. Variations in key characteristics of the kernels and coatings are reflected in measured particle layer properties from AGR-1 and AGR-2. UCO TRISO-coated fuel particles that satisfy the parameter envelope defined by these measured particle layer properties can be relied upon to provide satisfactory performance.
3. Aggregate AGR-1 and AGR-2 fission product release data and fuel failure fractions, as summarized in this report, can be used to support licensing of reactors employing UCO TRISO-coated fuel particles that satisfy the parameter envelope defined by measured particle layer properties from AGR-1 and AGR-2.

Keywords
Advanced nuclear fuel
Advanced nuclear technology
Advanced reactors
High temperature reactor (HTR)
Tristructural isotropic (TRISO)
Uranium oxycarbide (UCO)

EXECUTIVE SUMMARY

Deliverable Number: 3002019978

Product Type: Technical Report

Product Title: Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO)-Coated Particle Fuel Performance—Topical Report EPRI-AR-1(NP)-A

PRIMARY AUDIENCE: Developers of high-temperature reactor (HTR) designs that use uranium oxycarbide (UCO)-based tristructural isotropic (TRISO)-coated particle fuel and regulators involved in design certification or licensing of HTRs utilizing TRISO fuel

SECONDARY AUDIENCE: Future owner-operators and other stakeholders interested in the technical basis for TRISO fuel performance

KEY RESEARCH QUESTION

Most contemporary gas- and molten salt-cooled HTR designs rely on the performance of TRISO fuel particles embedded in prismatic blocks or pebble fuel; application to other reactor designs is possible and likely. EPRI has taken part in a collaborative effort involving the U.S. Department of Energy (DOE), Idaho National Laboratory, HTR developers, fuel suppliers, and other industry stakeholders. As part of its involvement, EPRI has developed a topical report on UCO TRISO-coated particle fuel performance to document key data and results from the first two phases of testing in the U.S. DOE Advanced Gas Reactor Fuel Development and Qualification (AGR) Program.

RESEARCH OVERVIEW

Modern TRISO particle fuel technology is the product of coated particle fuel development spanning many countries over a half-century period. DOE launched the AGR program in 2002 to establish the ability to manufacture high-quality TRISO fuel in the United States and demonstrate its performance. The first two phases of fuel irradiation testing, designated AGR-1 and AGR-2, provide data demonstrating the adequate performance of TRISO fuel during irradiation and in post-irradiation high-temperature accident safety tests. While most of these data have been or will be published in the public domain, the data have not been assembled in a concise format for efficient regulatory review and referencing by advanced reactor developers—a situation this topical report addresses.

KEY FINDINGS

- Testing of UCO TRISO-coated fuel particles in AGR-1 and AGR-2 constitutes a performance demonstration of these particle designs over a range of normal operating and off-normal accident conditions. Therefore, the testing provides a foundational basis for use of these particle designs in the fuel elements of TRISO-fueled HTR designs (that is, designs with pebble or prismatic fuel and helium or salt coolant).
- The kernels and coatings of the UCO TRISO-coated fuel particles tested in AGR-1 and AGR-2 exhibited property variations and were fabricated under different conditions and at different scales, with remarkably similar excellent irradiation and accident safety performance. Variations in key characteristics of the kernels and coatings are reflected in measured particle layer properties from AGR-1 and AGR-2. UCO TRISO-coated fuel particles that satisfy the parameter envelope defined by these measured particle layer properties can be relied upon to provide satisfactory performance.

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

WHY THIS MATTERS

This report consolidates foundational fuel performance data and results from the AGR-1 and AGR-2 tests in a form reviewed by the U.S. Nuclear Regulatory Commission (NRC) as documented in its August 2020 safety evaluation (SE). Designers and Developers can reference the approved topical report to address UCO TRISO fuel performance. Accordingly, this report "locks-in" existing fuel performance data and results in a form that can increase the efficiency of the safety review process for design certification and license applications in the United States and internationally.

HOW TO APPLY RESULTS

The results documented in this topical report provide a consolidated reference on the performance of UCO TRISO-coated fuel particles demonstrated in AGR-1 and AGR-2 testing. Coupled with the NRC review and approval—as documented in the August 2020 SE—this topical report should increase regulatory certainty of HTR design certification and licensing efforts by providing early acceptance of foundational information on UCO TRISO fuel particle performance.

LEARNING AND ENGAGEMENT OPPORTUNITIES

- EPRI has established an Advanced Reactor Technical Advisory Group (TAG) and a public facing Stakeholder Forum under the Advanced Nuclear Technology Program to provide channels for exchanging information and obtaining input on the direction and nature of EPRI's strategic focus on advanced reactor technology.
- EPRI continues to seek and welcome collaborative research and development (R&D) opportunities that support commercialization of advanced nuclear technology, including consolidation and publication of R&D results that facilitate reactor developer efforts to pursue design certification as well as licensing and associated regulatory review of those designs.

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PROGRAM: Advanced Nuclear Technology, P41.08.01, Technology Innovation, and Nuclear Power, P41

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ACRONYMS

3-D	three-dimensional
ACRS	Advisory Committee on Reactor Safeguards
ADUN	acid-deficient uranyl nitrate (solution)
AGR	Advanced Gas Reactor (Fuel Development and Qualification Program)
AOO	anticipated operational occurrence
ARDC	advanced reactor design criteria
ART	Advanced Reactor Technologies
ATR	Advanced Test Reactor (United States)
AVR	Arbeitsgemeinschaft Versuchsreaktor (Germany)
B&W	Babcock and Wilcox, Inc.
BAF	Bacon Anisotropy Factor
BISO	bistructural isotropic
BWXT	BWX Technologies, Inc. (also known as, Babcock and Wilcox or B&W)
CCCTF	Core Conduction Cooldown Test Facility (ORNL)
DLBL	deconsolidation-leach-burn-leach
DOE	U.S. Department of Energy
DTF	designed to fail
EAB	exclusion area boundary
EFPD	effective full-power day
EPA	U.S. Environmental Protection Agency
EPRI	Electric Power Research Institute
FACS	Fuel Accident Condition Simulator (INL)
FHR	fluoride salt-cooled high-temperature reactor
FIMA	fissions per initial metal atom
FPMS	fission product monitoring system

FRJ 2 DIDO	Forschungszentrum Jülich Research Reactor (Germany)
FSV	Fort St. Vrain (United States)
GA	General Atomics
GDC	general design criteria
GT-MHR	gas turbine modular helium reactor
HEU	highly enriched uranium
HFIR	high-flux isotope reactor (United States)
HFR	High Flux Reactor (Netherlands)
HM	heavy-metal
HMTA	hexamethylenetetramine
HPGe	high-purity germanium
HTGR	high-temperature gas-cooled reactor
HTR	high-temperature reactor
HTR-10	10-MW High-Temperature Gas-Cooled Reactor (China)
HTR-MODUL	200 MWth High Temperature Gas-Cooled Reactor (Interatom/Siemens/Framatome)
HTR-PM	High-Temperature Reactor-Pebble-bed Modular (China)
HTTR	High-Temperature Test Reactor (Japan)
IAEA	International Atomic Energy Agency
INET	Institute of Nuclear and New Energy Technology (China)
INL	Idaho National Laboratory (United States)
IPyC	inner pyrolytic carbon
JAERI	Japan Atomic Energy Research Institute (now Japan Atomic Energy Agency JAEA)
KüFA	KühlFinger-Apparatur (Cold Finger Apparatus)
LBE	licensing basis events
LEU	low-enriched uranium
LHTGR	large high-temperature gas-cooled reactor
LWR	light water reactor
MCNP	Monte Carlo N Particle
MeV	million electron volts
MHTGR	modular high-temperature gas-cooled reactor

MTHM	metric tons of heavy metal
MTR	materials test reactor
MTS	methyltrichlorosilane
NaI(Tl)	thallium-activated, sodium iodide (scintillation detector)
NEFT	northeast flux trap
NGNP	Next Generation Nuclear Plant
Non-LWR	non-light-water reactor
NP-MHTGR	New Production Modular High-Temperature Gas-Cooled Reactor
NPR	New Production Reactor
NRC	U.S. Nuclear Regulatory Commission
NTTF	Near-Term Task Force
NUREG	publication prepared by the NRC staff
OPyC	outer pyrolytic carbon
ORNL	Oak Ridge National Laboratory
PAG	Protective Action Guide
PBMR	Pebble-Bed Modular Reactor (South Africa)
PDC	principal design criteria
PIE	post-irradiation examination
PIRT	phenomena identification and ranking table
PSER	pre-application safety evaluation report
PyC	pyrolytic carbon (also pyrocarbon)
R&D	research and development
R/B	release-rate-to-birth-rate ratio
RG	regulatory guide
SARRDL	specified acceptable system radionuclide release design limits
SECY	NRC Commission papers
SFR	sodium-cooled fast reactor
SiC	silicon carbide
SSC	structures, systems and components
SST	stainless steel
STP	standard temperature and pressure

TAVA	time-average volume-average
TC	thermocouple
THTR	Thorium High-Temperature Reactor (Germany)
TRISO	tristructural isotropic
U.S.	United States
UC_x	uranium carbide
UCO	uranium oxycarbide
UO_2	uranium dioxide

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1
INTRODUCTION

Nuclear fuel qualification represents one of the longest-lead items for commercializing a new reactor technology. While fuel forms and operating conditions vary widely across the many advanced reactor designs under consideration and development, many high-temperature reactor (HTR) concepts use tristructural isotropic (TRISO)-coated particles as the basis for their fuel designs, including high-temperature gas-cooled reactors (HTGRs) and fluoride salt-cooled high-temperature reactors (FHRs).

A wide variety of HTR designs that would use TRISO-coated particle fuel in a carbonaceous matrix can operate safely under realistic operating and accident scenarios provided the time-at-temperature of the particles remains below fission product release thresholds. Experimental evidence shows that if these thresholds are not exceeded, a level of fuel performance and fission product retention is achieved such that the radioactive source term emitted from the plant will be lower by orders of magnitude than other reactor types. In the United States, siting of the plant near population centers and co-location with industrial users of process heat requires compliance of releases at or near the site boundary with U.S. Environmental Protection Agency (EPA) Protective Action Guides (PAG) for offsite dose. This enables a graded approach to emergency planning and the potential elimination of the need for evacuation and sheltering beyond the site boundary. However, achieving this level of performance is predicated on the fabrication of coated-particle fuel that demonstrates excellent performance under anticipated operating and accident conditions.

The U.S. Department of Energy initiated the Advanced Gas Reactor Fuel Development and Qualification (AGR) Program in 2002 to establish the capability in the U.S. to fabricate high-quality TRISO fuel and to demonstrate fuel performance. The AGR program to date has focused on manufacturing and testing the fuel design for HTR concepts using the most recent gas turbine modular helium reactor fuel product specification as a starting point [1]. Irradiation, safety testing, and post-irradiation examination (PIE) plans support fuel development and qualification in an integrated manner. The AGR program consists of four testing campaigns; AGR-1, AGR-2, AGR-3/4 and AGR-5/6/7. The first two fuel irradiation tests in the program, designated AGR-1 and AGR-2, demonstrated uranium oxycarbide (UCO)[1] fuel performance during irradiation and during post-irradiation high-temperature accident safety tests. **This topical report covers the foundational fuel performance testing from the AGR-1 and AGR-2 tests and the irradiation, safety testing and PIE results to date.**

[1] Uranium oxycarbide as used here is a short-hand term to denote a mixture of uranium dioxide (UO_2) and uranium carbide (UC_x), the two phases present in the kernel.

1.1 Report Scope and Purpose

This report provides the technical bases (that is, particle design, irradiation, and accident testing results) that demonstrate the functional performance of UCO TRISO-coated particles so these particles can be used by a variety of high-temperature reactor developers in their designs. This report addresses UCO fuel performance only. Any information related to uranium dioxide (UO_2) fuel is provided for context and comparison purposes only.[2]

On May 31, 2019, the original version of this report[3] was submitted to the U.S. Nuclear Regulatory Commission (NRC) as a topical report for formal review and approval documented with the issuance of a safety evaluation (SE). NRC Office of Nuclear Reactor Regulation Office Instruction LIC-500, Topical Report Process [2], defines a topical report as a stand-alone report containing technical information about a nuclear power plant safety topic. Further, a topical report provides the technical basis for a licensing action.

Topical reports are reviewed by the NRC staff with the intent of maximizing their scope of applicability consistent with current standards for licensing actions, compliance with the applicable regulations, and reasonable assurance the health and safety of the public will not be adversely affected. Topical reports improve the efficiency of the licensing process by allowing the staff to review proposed methodologies, designs, operational requirements, or other safety-related subjects on a generic basis, so they may be implemented by reference by multiple U.S. licensees once determined to be acceptable for use and verified by the NRC staff. By reviewing this information as a topical report, the NRC can reduce the review time for the technical bases by allowing applicants to reference the topical report and associated safety evaluation, rather than submitting it for review and approval on each application.

The review of the information provided in this topical report supports HTR developers and other stakeholders by:

- Providing early acceptance and resolution of technical information and foundational information for industry to move forward with a degree of design and regulatory certainty
- Identifying technology neutral open issues that might be resolved generically from subsequent AGR-3/4 or AGR-5/6/7 tests in subsequent topical reports or applications
- Identifying technology specific open issues that can be resolved in subsequent topical reports or applications
- Progressing fuel performance reviews in parallel with ongoing efforts on source term, functional containment performance, and the development/review of NEI-18-04 with respect to licensing basis events (LBE), structures, systems, and components (SSC) classification and defense in depth

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[2] While some limited work on UO_2 was included in AGR-2 as part of an international collaboration under the auspices of the Generation IV International Forum, the AGR program is focused on UCO TRISO-coated particle fuel. AGR UO_2 fuel performance is included in this report for context and background.

[3] *Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO) Coated Particle Fuel Performance: Topical Report EPRI-AR-1(NP)*. EPRI, Palo Alto, CA: May 2019. 3002015750. Transmittal dated May 31, 2019 documented as NRC ADAMS Accession No. ML19155A173. Docket No. 99902021.

- Providing data on fuel performance and fission product release that can be utilized as part of a computational code verification and validation effort

1.2 Report Content and Structure

The substantive content of this report is adapted from material prepared by INL as part of a collaborative project to develop and submit a topical report on UCO TRISO-coated particle fuel performance based on available results from the AGR-1 and AGR-2 campaigns [3].[4] The report content is organized and presented in the following manner:

- Section 2 provides an overview of the TRISO-related NRC Regulatory Bases, including a description of how this topical report fits into overall TRISO-fueled plant licensing strategies.
- Section 3 summarizes the background information for the basis of TRISO-coated particle fuel technology resulting from decades of development of TRISO-coated fuel particles in the United States.
- Section 4 introduces the concept of fission product retention for reactor systems that use TRISO-coated particle fuel and presents the basis for the particle design and performance used in the AGR program and provides representative levels of fuel performance requirements necessary to implement such an approach.
- Section 5 provides a brief overview of the AGR program, including the different program elements and the four fuel irradiation campaigns around which the program is structured. Fabrication of the AGR fuel is described in Section 5.3.
- Section 6 provides the irradiation response of fuel particles in the AGR-1 and AGR-2 campaigns.
- Section 7 presents follow-on safety test performance and PIE data for AGR-1 and AGR-2.
- Section 8 provides a summary of the report, including the key conclusions drawn from this work in regard to U.S. UCO TRISO fuel performance.
- Appendix A provides an overview of the regulatory history for the U.S. related to TRISO fuel.
- Appendix B provides an overview of the international TRISO-coated particle fuel experience base.
- Appendix C provides information from the AGR-1 and AGR-2 fuel specifications.
- Appendix D documents licensing correspondence associated with the NRC review of the topical report.

[4] Under a collaborative project jointly funded by EPRI and DOE, the technical content presented in Sections 2–7 and Appendices A and B was compiled and prepared specifically for this report with the assistance of Idaho National Laboratory (INL), Battelle Energy Alliance, LLC (BEA), under contract DE-AC07-05ID14517 with the U.S. Department of Energy. This content is also documented in INL/LTD-18-46060 Rev. 0, *Technical Bases for the Performance Demonstration of TRISO-coated UCO Fuel Particles* [3] and has been derived from other INL/BEA reports and results.

1.3 Key Conclusions for NRC Review and Approval

EPRI requested NRC review of AGR-1 and AGR-2 data and analyses documented in Sections 5-7 of this topical report[5] and sought NRC approval of the following three conclusions presented in Section 8:

1. Testing of UCO TRISO-coated fuel particles in AGR-1 and AGR-2 constitutes a performance demonstration of these particle designs over a range of normal operating and off-normal accident conditions. Therefore, the testing provides a foundational basis for use of these particle designs in the fuel elements of TRISO-fueled HTR designs (that is, designs with pebble or prismatic fuel and helium or salt coolant).
2. The kernels and coatings of the UCO TRISO-coated fuel particles tested in AGR-1 and AGR-2 exhibited property variations and were fabricated under different conditions and at different scales, with remarkably similar excellent irradiation and accident safety performance results. The ranges of those variations in key characteristics of the kernels and coatings are reflected in measured particle layer properties provided in Table 5-5 from AGR-1 and AGR-2. UCO TRISO-coated fuel particles that satisfy the parameter envelope defined by these measured particle layer properties in Table 5-5 can be relied on to provide satisfactory performance.
3. Aggregate AGR-1 and AGR-2 fission product release data and fuel failure fractions, as summarized in this report, can be used to support licensing of reactors employing UCO TRISO-coated fuel particles that satisfy the parameter envelope defined by measured particle layer properties in Table 5-5 from AGR-1 and AGR-2.

[5] Sections 5–7 document AGR program results and analyses that represent the core scope of this topical report and support the conclusions of this report. Sections 1–4 and Appendices A and B are included as historical background and context only. The U.S. TRISO fuel qualification program and the conclusions of this topical report do not rely upon historical pre-AGR data.

2
U.S. REGULATORY BASES

2.1 Prior NRC HTGR TRISO-Related Interactions

In 2005, the U.S. Department of Energy (DOE) established the Next Generation Nuclear Plant (NGNP) project at Idaho National Laboratory (INL) to support near-term commercial deployment of a HTGR technology demonstration plant. A key part of the project was the development of a regulatory framework supportive of commercial HTGR deployment. These activities were closely coordinated with NRC staff and focused on adapting existing nuclear power plant regulatory requirements to the needs of NGNP licensing. DOE and NRC jointly formulated the approach for this licensing structure and communicated this approach to Congress in 2008.

Under the NGNP project, HTGR licensing precedents and NRC regulations were examined systematically as they relate to the HTGR safety case and associated plant design goals. NRC staff coordinated the scope of this examination and reviewed the results. In 2009, this information was used to develop a strategic implementation plan [4] for establishing the regulatory basis necessary to complete and submit an HTGR license application to NRC. The plan focused on key elements of plant safety design and licensing and included:

- Developing the basis for establishing a mechanistic radiological source term based primarily on particle fuel design and available qualification testing results
- Preventing and mitigating the release of the radiological source terms to the environment, including methods for the structured and comprehensive identification of licensing basis event sequences along with establishing multiple radionuclide release barriers

The design and licensing strategy of the NGNP centered on radionuclide retention capabilities of TRISO particle fuel. It also relied less on other barriers for limiting offsite releases of radionuclides compared to historical light water reactor (LWR) technology.

In July 2014, the NRC issued a report summarizing the results of these regulatory framework interactions with the NGNP. Important outcomes identified in that NRC staff report [5] included:

- General agreement was expressed with the proposed HTGR performance standard concerning HTGR functional containment. The functional containment approach limits radionuclide releases to the environment by emphasizing retention of radionuclides at their source in the fuel rather than allowing significant fuel particle failures and relying upon other external barriers to provide compliance with identified top-level regulatory dose acceptance criteria.
- The INL AGR Fuel Development and Qualification Program was determined to be reasonably complete within a context of pre-prototype fuel testing. Early fuel test results showed promise in demonstrating much of the desired retention capabilities of the TRISO particle fuel.

2.2 Current NRC Regulatory Framework

The NRC conducts its reactor licensing activities through a combination of regulatory requirements and guidance. The applicable regulatory requirements are found in Chapter I of Title 10, "Energy," of the *Code of Federal Regulations*, Parts 1 through 199. Regulatory guidance provides additional detailed information on specific acceptable means to meet the requirements in regulation. Guidance exists in several forms, including: Regulatory Guides (RGs), interim staff guidance, standard review plans, publications prepared by the NRC staff (NUREGs), review standards, and Commission policy statements. Appendix A summarizes these regulatory and guidance documents related to TRISO fuel. These regulatory requirements and guidance represent the entirety of the regulatory framework an applicant should consider when preparing an application for review by the NRC.

Establishing principal design criteria (PDC) for a reactor is a key part of the NRC's regulatory framework. The general design criteria (GDC) contained in Appendix A to 10 CFR Part 50 [6] were developed specifically for LWRs and provide minimum requirements for PDC, which establish the necessary design, fabrication, construction, testing, and performance requirements for SSCs which are important to safety, that is, SSCs "that provide reasonable assurance that the nuclear power plant can be operated without undue risk to the health and safety of the public."

The GDC also provide guidance in establishing the PDC for non-light-water reactors (non-LWRs). The PDC serve as the fundamental criteria for the NRC staff when reviewing the SSCs that make up a nuclear power plant design particularly when assessing the performance of their intended safety functions in design basis events postulated to occur during normal operations, anticipated operational occurrences (AOO), and postulated accidents. All production and utilization facilities licensed under 10 CFR Part 50, including both LWRs and non-LWRs, are required to describe PDC in their preliminary safety analysis report supporting a construction permit application as described in 10 CFR 50.34(a)(3). All applicants for a standard design certification are also required to describe PDC in their final safety analysis report as described in 10 CFR 52.47(a)(3).

In April 2018, the NRC issued RG 1.232 "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors," which provides guidance for how the GDC in 10 CFR 50 Appendix A may be adapted to develop PDC for non-LWR designs [7]. In addition, RG 1.232 provides guidance for adapting the LWR GDC for modular HTGRs and sodium-cooled fast reactors (SFRs). RG 1.232 guidance may be used to develop all or part of a design's PDC and users may choose among the Advanced Reactor design criteria (ARDC), modular HTGR design criteria (MHTGR-DC), or SFR design criteria (SFR-DC) to develop their PDC after considering the underlying safety basis a given criterion and evaluating the RG's rationale for the adaptation.

The work to develop and issue this regulatory guidance provides key regulatory underpinning for the path forward on advanced reactors. Specifically, MHTGR-DC 10 and MHTGR-DC 16, provide a model for evaluation of TRISO fuel performance in combination with plant systems performance and functional containment performance to achieve the overall radiological dose criteria. This work on particle fuel design and performance testing supports development of the basis for establishing a mechanistic radiological source term.

MHTGR-DC 10, *Reactor Design*, provides guidance related to acceptable system radionuclide releases. Other ARDC that pertain to the reactor core (that is, MHTGR-DC 11, 12, 13, and 26) do not directly pertain to the performance of the TRISO-coated particle fuel. MHTGR-DC 10, states [7]:

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

RG 1.232 includes the following rationale for MHTGR-DC 10 documenting the basis for wording changes from the original LWR GDC:

- "the concept of specified acceptable fuel design limits, which prevent additional fuel failures during anticipated operational occurrences (AOOs), has been replaced with that of the specified acceptable system radionuclide release design limits (SARRDLs), which limits the amount of radionuclide inventory that is released by the system under normal and AOO conditions." Design features within the reactor system must ensure the SARRDLs are not exceeded during normal operations and AOOs.
- The TRISO fuel used in the MHTGR design is the primary fission product barrier and is expected to have a very low incremental fission product release during AOOs.
- The SARRDLs will be established to ensure that the most limiting license-basis event does not exceed the siting regulatory dose limits criteria at the exclusion area boundary (EAB) and low-population zone (LPZ), and to ensure the 10 CFR 20.1301 annualized dose limits to the public are not exceeded at the EAB for normal operation and AOOs.
- The concept of the TRISO fuel being the primary fission product barrier is intertwined with the concept of a functional containment for MHTGR technologies. See the rationale for MHTGR-DC 16 for further information on the Commission's current position.

MHTGR-DC 16, *Containment Design*, provides guidance for a functional containment design, which relies on the use of multiple barriers to control the release of radioactivity. MHTGR-DC 16 states [7]:

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

RG 1.232 includes the following rationale for MHTGR-DC 16 documenting the basis for wording changes from the original LWR GDC, which include [7]:

- The term 'functional containment' is applicable to advanced non-LWRs without a pressure retaining containment structure. A functional containment can be defined as "a barrier, or set of barriers taken together, that effectively limit the physical transport and release of radionuclides to the environment across a full range of normal operating conditions, AOOs, and accident conditions."

- "The NRC staff has brought the issue of functional containment to the Commission, and the Commission has found it generally acceptable."
- "The NRC staff also provided feedback to the DOE on this issue as part of the NGNP project, (*see Appendix A to this document*). ... the area on functional containment and fuel development and qualification noted that "...approval of the proposed approach to functional containment for the MHTGR concept, with its emphasis on passive safety features and radionuclide retention within the fuel over a broad spectrum of off-normal conditions, would necessitate that the required fuel particle performance capabilities be demonstrated with a high degree of certainty."

Figure 2-1 below illustrates how this topical report fits conceptually into the broader context of technology inclusive TRISO fuel performance, future AGR program data, manufacturing specifications and evaluation, design specific systems evaluation, functional containment evaluation, and design specific demonstration of achieving acceptable dose criteria. This figure is intended to illustrate where and how this topical report provides valuable foundational information and finality to industry and the NRC. It is not intended to capture all of the steps in the future review process for the ultimate licensing of a plant.

As highlighted in the figure, this topical report addresses only the performance data obtained in the AGR-1 and AGR-2 tests. While limited in scope, these data are foundational to the design and licensing of reactors using TRISO fuel. This topical report provides TRISO fuel performance data from the AGR-1 and AGR-2 tests for use by future applicants during design of the plants and for use by NRC staff in accepting the design inputs and test data for fuel performance. The results presented here demonstrate the excellent performance of TRISO-coated fuel particles under normal and postulated accident conditions.

The completion of future AGR tests (discussed in Section 5 below) will provide additional information on statistical performance testing, fission product transport, and fuel performance margin tests. These data will also be important to future applicants and to NRC Staff for completion of safety evaluations. Applicants will utilize these data to formulate their fuel licensing case. The figure illustrates options that could include for example; an amendment to this report, a future stand-alone topical report, or use of test reports to support a future application and review.

The figure also indicates that future applicants will be required to develop LBE, demonstrate that the specific reactor design is within the range of applicability for the TRISO particle performance data, incorporate this information into the system design, establish SARRDLs, establish the functional containment design, and demonstrate that acceptable dose criteria are achieved for the plant.

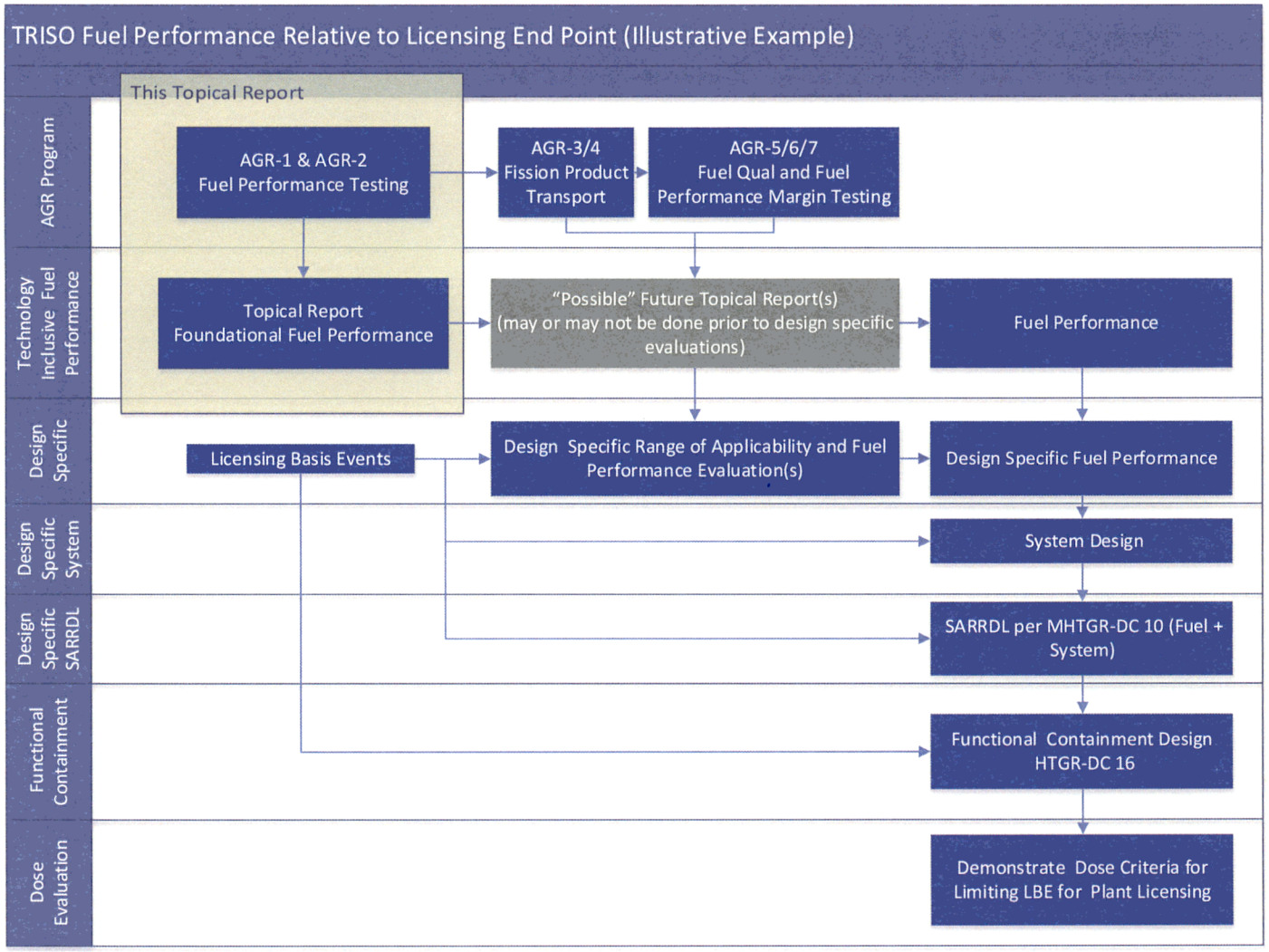

Figure 2-1
TRISO fuel performance relative to licensing end point (illustrative example)

3
TRISO-COATED PARTICLE FUEL EXPERIENCE BASE

This section reviews the existing experience base supporting the development, qualification, and production of TRISO-coated particle fuel. A broad base of experience encompassing a range of coated particle designs and service conditions provides a general understanding of the important phenomena associated with particle fabrication and performance and has served to identify potential fuel failure mechanisms. This experience yields a common internationally-recognized set of particle design features, which, in combination with restrictions on service conditions, mitigate or eliminate failure mechanisms.

The coated particles must be designed and fabricated to remain intact and retain radionuclides with a high level of effectiveness over the range of conditions that could be encountered in normal operation and under accident conditions. Historic modular gas reactor design concepts have been developed to limit the fuel service conditions (for example, burnup, fast fluence, temperature) to a range consistent with the performance capabilities of the fuel. The particles must be able to accommodate the following effects:

- Fission-induced changes in the kernel: production of a wide range of fission-product[6] isotopes, lattice dislocations by fission product recoil, kernel swelling due to solid and gaseous fission products, liberation of oxygen from fissioning of UO_2 molecules
- High-energy neutron-induced changes in material microstructure: anisotropic shrinkage and/or expansion in pyrocarbon layers, reductions in silicon carbide (SiC) layer strength
- Buildup of pressure within the particles: release of noble gas fission products from the kernel, production of CO and CO_2 from reaction of excess oxygen with buffer material, mainly in the case of UO_2 kernels
- Redistribution of fission products within the particle and chemical reactions with particle layers: chemical attack of the SiC layer and migration of the kernel within the particle

The last three effects are time and temperature dependent with a wide range of rate constants.

Particle physical characteristics established to meet anticipated performance requirements include dimensions (mean and variation), densities, pyrocarbon anisotropy, and defect levels. Rigorous statistically based procedures are used to characterize this fuel.

Experience with manufacturing coated-particle fuel has demonstrated the feasibility of producing large quantities of coated-particle fuel with low as-manufactured defect levels, approaching defect fractions of 10^{-5}. This capability was first demonstrated in Germany with the production of reload fuel batches for the Arbeitsgemeinschaft Versuchsreaktor (AVR) and subsequently

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[6] The term "fission product" here is used broadly to include isotopes that are produced as a result of fission processes (that is, direct fission products or isotopes that result from the radioactive decay of direct fission products) and isotopes resulting primarily from neutron activation of fission products (important examples include ^{110m}Ag and ^{134}Cs).

confirmed in fuel production campaigns in Japan for the High-Temperature Test Reactor (HTTR) first core and in China for the 10-MW High-Temperature Gas-Cooled Reactor (HTR-10) first core. Laboratory-scale production of high-quality fuel has also been demonstrated in Russia, South Africa, and the United States.

Appendix B summarizes the broad international experience with coated-particle fuel fabrication and performance covering a wide range of particle designs and material properties explored in the evolution toward the LEU TRISO particle under common development today. It also addresses the failure mechanisms that have been identified from this experience and the common particle design elements that have emerged.

The extensive international experience highlighted here and described in more detail in Appendix B includes particle designs exhibiting a wide variety of kernel properties. The kernel of the coated particle is substantially decoupled from the dense pyrocarbon and SiC layers by the low-density-carbon buffer layer. Thus, the experience generally applies to low-enriched uranium (LEU) UCO fuel from the standpoint of dense pyrocarbon and SiC-layer design and performance.

Section 4.2 describes the common elements of coated-particle designs that evolved from this broad experience and are under development. Section 4.3 addresses the potential particle failure mechanisms that were identified from the broad experience discussed in Section 3.1. These design elements, in combination with limitations established by the reactor designs on fuel-service conditions (for example, temperature, burnup, and fast fluence) under normal operation and accident conditions, effectively exclude most of the failure mechanisms and limit the remaining mechanisms to a very small fraction of the particles within a small fraction of the core.

3.1 Particle Development Experience

3.1.1 General Experience and Coated Particle Evolution

Coated particles start with a spherical kernel of fissile or fertile material surrounded by one or more refractory coatings. By the early 1960s, coated-particle fuel development for resinated graphite-moderated helium-cooled HTGRs was well under way in the United Kingdom in support of the DRAGON research reactor [8], in the U.S. in support of the Peach Bottom Unit 1 prototype power reactor [9], and in Germany in support of the AVR research and power reactor [10]. AVR fuel loadings evolved through many designs in the course of over two decades of plant operation, including the LEU TRISO design discussed in Section 4.2. As Figure 3-1 illustrates, coated particle designs for these early reactors varied considerably.

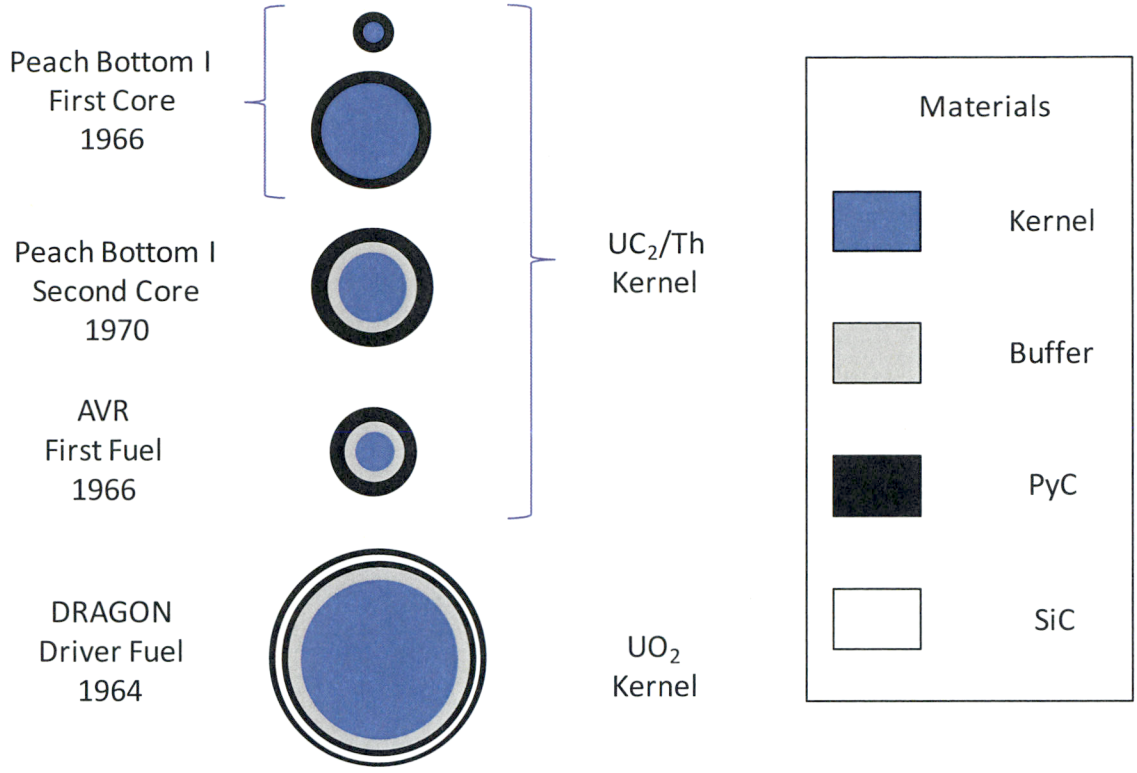

Figure 3-1
Early coated-particle designs
Courtesy of Idaho National Laboratory and used with permission of Battelle Energy Alliance, LLC

As described in Appendix B, coated-particle fuel development programs have also been conducted in France, Russia, Japan, China, South Africa, and South Korea. The development of coated-particle fuel technology for both the pebble-bed and prismatic designs has drawn from an extensive international background of coated-particle fuel fabrication and testing experience spanning more than 50 years and covering a broad range of parameters:

- Kernel characteristics:
  - Diameter – 100 to 800 μm
  - Fissile/fertile materials – uranium, thorium, plutonium (mixed and unmixed)
  - Chemical forms – oxide, carbide, oxycarbide
  - Enrichment – ranging from natural to high-enriched uranium (HEU) and plutonium
- Coating characteristics:
  - Bistructural isotropic (BISO) – variations in buffer and pyrolytic carbon (PyC) coating thicknesses and properties
  - TRISO – variations in buffer, PyC and SiC (or zirconium carbide) thicknesses and properties

- Fuel forms:
  - Spheres – multiple geometries and fabrication methods
  - Compacts – cylindrical and annular shapes with variations in particle packing fractions and fabrication methods
- Irradiation facilities:
  - Material Test Reactors – High Flux Reactor (HFR, Netherlands), Forschungszentrum Jülich Research Reactor (FRJ 2 DIDO, Germany), IVV-2M (Russia), Siloe (France), R2 (Sweden), BR2 (Belgium), High-Flux Isotope Reactor (HFIR, United States) and Advanced Test Reactor (ATR, United States), with wide variations in neutron energy spectra and degree of irradiation acceleration
  - Research and Demonstration Reactors – DRAGON (United Kingdom), Peach Bottom I (United States), AVR (Germany), Fort St. Vrain (FSV, United States), Thorium High Temperature Reactor (THTR, Germany), HTTR (Japan), and HTR-10 (China)
- Irradiation and testing conditions:
  - Burnup – ranging from below 1% to above 70% fissions per initial metal atom (FIMA)
  - Fast fluence – ranging from below 1×10^{21} to above 10×10^{21} n/cm^2
  - Irradiation temperature – ranging from 600 to 1950°C
  - Accident simulation temperature – ranging from 1400 to 2500°C

A detailed understanding of the parameters and phenomena of importance in the fabrication and performance of coated-particle fuel has emerged from this broad range of experience and data. Extensive bilateral and multilateral international information exchanges facilitated the incorporation of this broad experience base into German and other modern coated-particle fuels.

A detailed review of U.S. and German experience and the relationship to fuel performance and fuel performance modeling is documented in a 2004 EPRI report [11]. The evolution of the German fuel design, arriving at the LEU UO_2 TRISO pressed sphere selected as a basis for the pebble-bed reactor concept, is summarized in a historical review of AVR operation [10]. A broader range of international experience, focused mainly on LEU TRISO fuel, was addressed in an International Atomic Energy Agency (IAEA) coordinated research project conducted in the 1990s [12]. A more recent coordinated research project on TRISO-coated particle fuel was conducted in the early 2000s [13], which included two key elements: (1) an international quality control round robin test campaign for measuring important attributes of TRISO-coated particles; and (2) an international fuel performance benchmarking exercise to compare international codes that model TRISO-coated particle fuel under both normal operation and postulated accident conditions.

One important outcome of this international experience and data has been the convergence on common LEU TRISO particle designs, as discussed in Section 4.2, exhibiting similar coating thicknesses and properties with variations in kernel diameter, enrichment, and composition (UO_2 and UCO), depending on specific service conditions and requirements.

3.1.2 Experience Prior to U.S. AGR Program

Experience prior to the AGR program with irradiation and safety testing of TRISO-coated UCO particles is discussed in this section.

3.1.2.1 Fabrication

In the 1960s and 1970s, a large-scale coated-particle fuel-fabrication facility was established at General Atomics (GA) in the United States to support the operation of the 115-MWth Peach Bottom Unit 1 (cylindrical annular fuel compacts containing BISO-coated $(Th,U)C_2$ fuel particles) and the 842-MWth FSV (prismatic fuel elements containing TRISO-coated $(Th,U)C_2$ fissile particles and TRISO-coated ThC_2 fertile particles) HTGRs [14]. Following the termination of FSV operations in 1989, the fuel fabrication facility was used for the fabrication of some fuel test articles and all the TRISO target test compacts for the U.S. DOE's New Production Modular High-Temperature Gas-Cooled Reactor (NP-MHTGR), one of the technologies being developed for the New Production Reactor (NPR) program. Following cancellation of the NPR program, the facility was decommissioned and dismantled, thus eliminating large-scale TRISO fuel fabrication capability in the United States.

High-density UCO kernels were irradiated in twelve irradiation test capsules in the United States and Germany. Three production lots of high-density UCO kernels supplied all U.S. irradiations. The first U.S. production lot of 350 μm-diameter UCO was manufactured at GA. Compacts and loose particles from this batch were irradiated in capsules HRB-14, 15A, 15B, 16, 17, and 18, and R2-KI3. A second production run of 350-μm-diameter UCO was made by this same process for capsule HRB-21. The fuel kernels for HRB-21 were coated with the TRISO-P coating (a particle design featuring a sacrificial overcoating of low-density PyC in a fluidized particle bed to increase crush strength and reduce coating failure during matrix injection). A third batch of high-density, 200-μm-diameter UCO was made at Babcock and Wilcox (B&W) by the internal gelation process for use in the NP-MHTGR capsules. Subsequently, BWXT prepared UCO kernels for the AGR program starting in 2003 and developed coatings for AGR-2 and later capsules in 2004 until recently. These coaters were pilot scale six-inch (152 mm) coaters. A more complete description of the fuel particles and the U.S. irradiation experiments is provided in Petti et al. [15].

3.1.2.2 Irradiation

The U.S. irradiation program is described in Petti et al. [15]. Important results are presented here on irradiation of UCO fuel in both U.S. and German experiments prior to the AGR program.

Historical performance of UCO fuel in the early U.S. irradiation tests [15] does not meet the irradiation-performance requirements for current prismatic HTGR designs, but for reasons that appear unrelated to the performance of the UCO kernel. Instead, the performance issues appear to result from defective SiC coatings, which were created during coating and/or compacting processes. Examination of UCO particles during the PIE of these capsules did not reveal any evidence of failure that could readily be attributed to the UCO kernels. The irradiations confirmed the UCO kernels retained lanthanide fission products in the kernels and suppressed kernel migration and formation of CO in that no evidence of kernel migration or of attack on the SiC by CO or lanthanide fission products was observed. UO_2 particles mixed with UCO particles in the same compact exhibited significant kernel migration, while no kernel migration was observed in the UCO particles.

Capsule HRB-21 and the NP-MHTGR capsules all contained TRISO-P UCO particles. GA attributed the high-coating failure in these capsules to the poor design of the TRISO-P coating system, that is, rapid shrinkage of low-density outer pyrolytic carbon (OPyC) layer caused by introduction of a seal coat on the conventional OPyC layer, and the properties of the inner pyrolytic carbon (IPyC) layer (high anisotropy), and not to the UCO kernel itself [15,16,17]. HRB-21 LEU UCO was irradiated to 22% FIMA. The three NP-MHTGR capsules containing 200-μm-diameter HEU UCO fuel particles were irradiated up to approximately 78% FIMA.

By contrast, the German capsule FRJ2-P24 irradiation of UCO under representative prismatic HTGR temperatures and burnup (but very low fast fluence) showed excellent fuel performance with respect to fission-gas retention. TRISO-coated 300-μm-diameter 20% enriched UCO particles formed into annular cylindrical fuel compacts were irradiated in this capsule. The UCO fuel achieved a burnup of up to 22% FIMA at a time-average temperature of about 1120°C with no in-service coating failures observed. No kernel migration or SiC corrosion because of fission product attack was reported by Borchardt et al. [18] and Bauer et al. [19].

In 1977, 5,354 fuel spheres (about 21% of the full AVR core) containing high-density TRISO-coated HEU UCO fuel kernels were inserted into the AVR. This was the first large-scale test of UCO in Germany. The fission-gas release in the AVR, as measured by the release-rate-to-birth-rate ratio (R/B[7]), remained at a level of $2–3 \times 10^{-5}$ $R/B^{85m}Kr$ while the UCO fuel spheres were under irradiation (similar to levels prior to UCO insertion). This provided a gross indication there was not extensive UCO particle failure. Given these R/B levels and the presence of the other fuel types in the core, a quantitative determination of the fuel performance was not possible. HEU fuel development was discontinued in Germany due to non-proliferation considerations. In 1982, the German HTGR program selected UO_2 for its reference fuel; consequently, no significant PIE or post-irradiation accident heating tests were performed on the HEU UCO fuel spheres irradiated in AVR.

Although the success of the German and Japanese fuel development programs (discussed in Appendix B) provides a high-level of confidence that TRISO fuel meeting prismatic HTR fuel performance requirements can be fabricated, this capability had not been demonstrated in the U.S. before DOE-sponsored commercial HTGR development ended in 1995. Consequently, DOE initiated the AGR program in 2002 to develop and qualify TRISO UCO fuel for HTRs to support future U.S. HTGR deployment.

[7] R/B is an indicator of initial fuel quality and fuel performance; it is defined as the ratio of the release rate (measured) over the birth rate (calculated) of short-lived fission gases that are released from exposed kernels (as a result of defective or failed coating layers) or dispersed uranium contamination outside of the coating layers. Fractional releases of short-lived fission gases can be expressed as R/B because the radioactive equilibrium is established relatively quickly in the fuel. Section 6.7 provides additional information on R/B ratios.

3.1.2.3 Safety Testing

While the German capsule FRJ2-P24 irradiation of UCO at representative prismatic HTGR temperatures and burnups showed excellent fuel performance with respect to retention of gaseous fission products, no post-irradiation simulated-accident heating tests were performed on the fuel from this capsule. In the U.S. program, fuel from irradiation tests HRB-15A and in HRB-15B was subjected to post-irradiation heating along with several other fuel types, including UO_2 and UC_2.

In one test series, 186 initially intact LEU UCO fuel particles from HRB-15A and HRB-15B were heated in temperature ramp and isothermal accident-simulation tests [20]. Krypton-85 release from the fuel was used to indicate total TRISO coating failure. The temperature ramp tests covered the range from ~1100°C to temperatures as high as approximately 2700°C, with heating rates in the range of ~19°C/h to ~190°C/h. The isothermal heating tests were conducted at 2050°C, 2200°C, and 2400°C. These temperatures and heating rates were representative of those expected in the large HTGR (LHTGR) designs under consideration at that time and are much higher than anticipated in HTR designs under consideration today. The test series also involved heating of other TRISO fuel types:

- HEU, LEU, and depleted UC_2
- ThO_2
- $(Th,U)O_2$
- LEU UO_2 and "UO_2*"[8]
- ThC_2 and $(Th,U)C_2$

Figure 3-2 (reproduced from reference 20) summarizes the results of thirty-hour ramp heating tests for the various LEU fuels and HEU UC_2. The primary mechanism for TRISO-coating failure and ^{85m}Kr release under the simulated LHTGR accident conditions was found to be thermal decomposition of the SiC layer, followed by either diffusion of fission products through the PyC layers or breakage of the PyC. Within the temperature range tested, fuel-particle performance was found to depend on the inherent thermal stability of the SiC coating layer and not to be dependent upon variations in burnup, fast neutron fluence, or kernel composition.

[8] UO_2* has a ZrC layer over the UO_2 kernel.

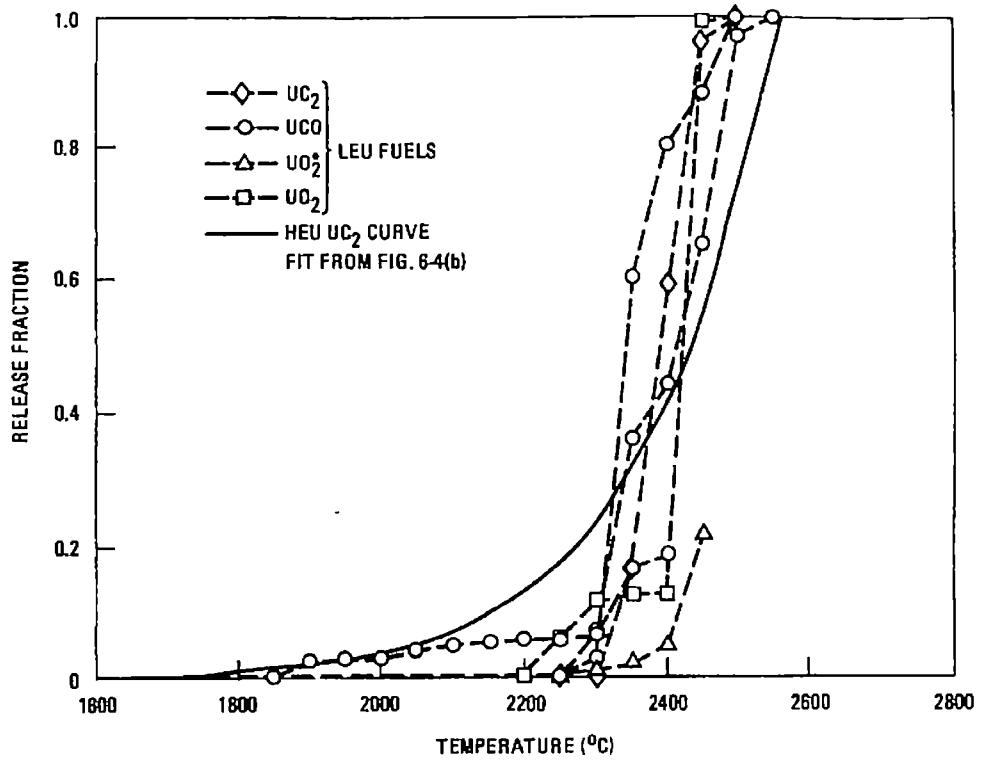

Figure 3-2
^{85m}Kr release results for ramp heating tests of candidate HTGR fuel types [20].
Courtesy of General Atomics; used with permission. All rights reserved.

Given the different chemical compositions of the fuel types, the similarity of the release profiles in Figure 3-2 indicates that performance of the fuel particles for the LHTGR accident conditions simulated in this heating-test series is independent of kernel composition and depends only on the TRISO coating. It is worth noting the temperatures associated with the LHTGR accident conditions are much higher than the temperatures during loss-of-forced-cooling accidents in the HTR designs being considered today.

In another heating-test series, 30 initially intact LEU UCO fuel particles irradiated in HRB-15B were heated isothermally for 10,000 hours at temperatures of 1200, 1350, or 1500°C (10 particles at each temperature) [19]. LEU UO_2, UC_2, and two variations of UO_2*[9] were also tested under the same conditions. With respect to the relative heating test performance of the UCO and UO_2 particles, the following differences were observed:

- At 1500°C, ^{154}Eu release started much earlier in the UCO fuel particles than the UO_2 particles, and the total ^{154}Eu release from the UCO particles (~50%) was considerably higher than from the UO_2 particles (~15%). The UCO particles also released ^{154}Eu at both 1200°C and 1350°C, but the amount released decreased significantly with decreasing temperature.[10]

[9] One version of UO_2* had a ZrC-coated UO_2 kernel encapsulated by a standard TRISO coating. The ZrC coating layer on the kernel had a thickness of about 10 microns. The other version of UO_2* used standard TRISO-coated UO_2 particles, except that ZrC was distributed within the buffer coating layer.

[10] Eu is significantly retained by the graphite fuel blocks, so the increased release of Eu isotopes from UCO fuel particles , relative to UO_2 fuel particles is not a significant issue for UCO fuel used in a prismatic HTGR.

The UO_2 particles did not release ^{154}Eu at 1200°C or 1350°C. (Similar results have been observed in AGR-1 and AGR-2 tests as discussed in Section 7.1.2.).

- At 1500°C, ^{110m}Ag release started much earlier in the UO_2 particles than the UCO particles, and the total ^{110m}Ag release from the UO_2 particles (~90%) was considerably higher than from the UCO particles (<10%).
- ^{137}Cs was released only at 1500°C and only from three of the 150 particles tested. Two of these were UO_2 particles. Diffusion through flawed, but intact, SiC layers was apparently responsible for the steadily increasing release from the two UO_2 particles. None of the UCO particles released ^{137}Cs at any temperature.

Although the above results indicate some differences in the accident-condition performance of UCO and UO_2 fuel particles, it is important to note there were substantial differences in the SiC coatings on these two types of particles, which likely influenced fission product retention. The SiC layer on the UCO particles is characterized as having a laminar microstructure and a density of 3.16 Mg/m^3; the SiC layer on the UO_2 particles is characterized as having a columnar microstructure and a density of 3.21 Mg/m^3 [21].

In an additional heating test of U.S. UCO fuel, fuel-compact-containing carbonaceous matrix body sections from irradiation test R2-K13 were heated in Germany [22]. These samples were heated to 2500°C, resulting in total failure of the SiC.

4
FISSION PRODUCT RETENTION, PARTICLE DESIGN, AND PERFORMANCE BASES

4.1 Fission Product Retention

High-temperature reactors possess design features that result in multiple barriers working together to attenuate the release of radionuclides. This concept is called "functional containment"[11] and encompasses a collection of design selections that, when taken together, ensure: (1) radionuclides are retained within multiple barriers arrayed in series, (with emphasis on retention at their source in the fuel); and (2) regulatory requirements and plant design goals for release of radionuclides are met (typically at the exclusion area boundary). The first three functional containment barriers consist of the fuel kernel, the fuel particle coatings, and the fuel matrix/material. For HTGRs, the fourth barrier is the helium pressure boundary. In the case of a fluoride salt-cooled high-temperature reactor (FHR), the salt coolant also acts as a barrier due to its ability to retain radionuclides. The reactor building serves as the final barrier.

Operational and design features of HTRs also play an important role in the functional containment concept of retaining radionuclides during normal and accident scenarios. The degree to which individual functional barriers are relied upon during a particular accident sequence is a design choice that considers tradeoffs between the required effectiveness of different barriers in a specific design approach [23]. Collectively these barriers operate to reduce fission product releases to very low levels during normal operations and under design basis events, including postulated accidents [24].

Successfully implementing a safety strategy based on functional containment will require:

- TRISO fuel that can be fabricated and characterized in a repeatable and consistent manner
- Fuel performance with very low in-service failures
- A mechanistic source term that can be calculated to the requisite level of accuracy for both normal and off-normal conditions

Historically, HTGR designers established fuel performance requirements that ensured offsite (plant boundary) dose limits would not be exceeded. Table 4-1 lists representative levels of allowable fuel defects and the allowable levels of in-service failures under normal operation and postulated core heatup accidents at 95% confidence. This information is based on the legacy MHTGR prismatic design and the 200 MWth High Temperature Gas-Cooled Reactor (HTR-MODUL) pebble-bed. These values are very similar despite differences in the design

[11] Functional containment, as defined by NRC Regulatory Guide 1.232: "a barrier, or set of barriers taken together, that effectively limit the physical transport and release of radionuclides to the environment across a full range of normal operating conditions, AOOs, and accident conditions" (see MHTGR-DC 16, *Containment Design*).

service conditions of the fuel (for example, burnup, fast fluence, temperature). While the actual values used in a particular HTGR or FHR design are at the discretion of the designer, the values presented here can be used as a metric when comparing the performance of UCO TRISO-coated particles against those fabricated and tested under the AGR program.

Table 4-1
Representative fuel defect levels and in-service failures for historic HTGR designs

	Modular HTGR Prismatic	HTR-MODUL Pebble
Manufacturing Defect Level		
Heavy Metal Contamination	2×10^{-5}	6×10^{-5}
SiC Defects	1×10^{-4}	
In-Service Performance Requirements		
Incremental Failures Normal Operation	2×10^{-4}	1.6×10^{-4}
Incremental Failures Core Heatup Accidents	6×10^{-4}	6.6×10^{-4}

4.2 Particle Design

The broad coated-particle fuel fabrication, irradiation, and testing experience discussed in Section 3.1 and Appendix B, combined with effective international information exchanges, has resulted in a consensus on basic coated-particle properties among ongoing fuel-development programs, as illustrated in Figure 4-1 and discussed below. The TRISO-coated particle is a spherical, layered composite. For the AGR program, it consists of a kernel of UCO surrounded by a porous carbon buffer layer that accommodates fission recoils preventing direct damage to the other coating layers and allows space for fission gases to accumulate. Surrounding the buffer layer is an IPyC layer, a SiC layer, and an OPyC layer. Historically, a broad range of TRISO particles have been fabricated and tested around the world, including: UO_2, $(U,Th)O_2$, UC_2, (U,Th)C, PuO_2 and UCO (see Appendix B for more detail). Some of the designs also incorporated fertile particles, that is, ThO_2 or ThC_2 and natural or depleted UO_2, as a part of a fissile-fertile fuel system.

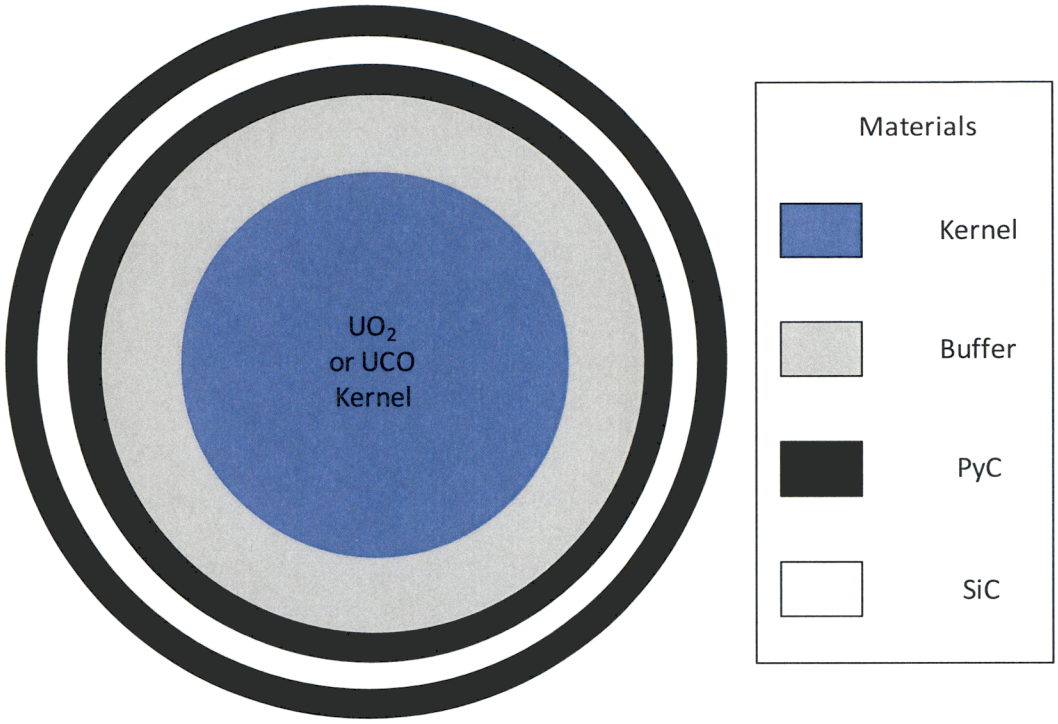

Figure 4-1
The international-consensus TRISO particle design
Courtesy of Idaho National Laboratory and used with permission of Battelle Energy Alliance, LLC

The coating layers of the TRISO fuel particles work synergistically to inhibit the release and migration of fission products from the fuel particle. The TRISO particles are embedded inside a carbonaceous matrix that provides a rigid structure, improves heat transfer and temperature uniformity, and retards migration of fission products that are not retained within the TRISO particles. This coated-particle design mitigates or eliminates the failure mechanisms discussed in Section 4.3 and incorporates the elements listed below. The mean coating thicknesses are sufficient to perform the required functions with allowance for the particle-to-particle variation in thickness resulting from the coating process.

4.2.1 Fuel Kernel

The spherical fuel kernel consists of high-assay, low-enriched (<20% ^{235}U) UO_2 or UCO. 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 by retention in the kernel and radioactive decay before release from the kernel.

UO_2 kernels perform effectively within the range of burnup and temperature gradients experienced in the German pebble-bed designs. Although some UO_2 kernels were fabricated as part of the AGR-2 campaign, the AGR program has focused efforts on characterizing and demonstrating the performance of UCO kernels. UCO kernels effectively limit the oxygen activity in the fuel, limiting the generation of CO and CO_2 and the associated kernel migration

and increased gas pressure in the particle. This allows higher burnup limits and thermal gradients associated with prismatic designs. The optimal kernel diameter is a function of enrichment and the related design burnup limits, with higher enrichment and burnup designs typically having smaller diameters.

The thermochemical basis for limited CO formation in UCO kernels is the oxidation of uranium carbide (UC_x) phases in response to the increasing oxygen potential in the UO_{2+x} phase as irradiation proceeds [25,26]. Past experimental measurements of CO formation in UO_2 + UC_x kernels indicate a drastic reduction compared to UO_2 [27]. In addition, both the historic UCO fuel irradiation testing database and the current AGR program results demonstrate the lower CO production based on the absence of any phenomena that are driven by CO pressure in the particles (for example, kernel migration or CO corrosion of the SiC layer).

4.2.2 Buffer Layer

The low-density (~50% of theoretical), porous PyC buffer coating layer protects the outer three layers by absorbing the kinetic energy of fission fragments ejected from the fuel kernel surface and providing space for the accumulation of gaseous fission products and carbon monoxide (in the case of UO_2 kernels). As a compressible material, it serves to mechanically 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 shrinks under irradiation as the kernel swells. The buffer layer is not considered a retentive layer for fission products, but fission gases and carbon monoxide do collect within the buffer pores. The buffer thickness is typically 90 to 100 μm.

4.2.3 Inner Pyrolytic Carbon Layer

The inner high-density (~85% of theoretical) isotropic layer of IPyC forms the second coating layer and the first load-bearing barrier against the pressure exerted by gaseous fission products and reaction products (CO, CO_2) within the fuel kernel and buffer layer. The IPyC layer also serves to protect the kernel from corrosive gases (HCl, Cl_2) liberated during the SiC coating process. Both the IPyC and OPyC layers retain gaseous fission products but become less effective in retaining metallic fission products at higher temperatures. The SiC occupies the surface-connected pores of the IPyC during deposition, thereby interlocking the two layers and providing extra mechanical support at the IPyC/SiC interface. The anisotropy of the IPyC layer is limited to control dimensional changes during irradiation where the IPyC and OPyC layers shrink at first, but may expand again if sufficiently high fast neutron dose levels are reached. Shrinkage of the IPyC layer during irradiation imparts a compressive load on the SiC layer. This reduces the maximum tensile hoop stress within the SiC, reducing the probability of in-pile particle failures. The IPyC thickness is typically 35 to 40 μm.

4.2.4 Silicon Carbide Layer

The SiC layer functions as the structural "skeleton" of the TRISO particle and is the third and the most important coating layer for fission product retention. Since the pyrocarbon layers become less effective in retaining metallic fission products at higher temperatures, the SiC layer acts as the principal barrier to the release of these elements from the coated particle. A high-density SiC with a non-columnar grain structure is considered the most effective for fission product retention.

The SiC layer also has sufficient strength to withstand internal pressure produced during irradiation. The coated particle structure and dimensional stability of the SiC layer under irradiation, combined with the irradiation-induced shrinkage of the IPyC and OPyC, results in the SiC layer being kept under compression during irradiation. This provides a high level of assurance the SiC layer will remain intact. The SiC thickness is typically ~35 µm.

4.2.5 Outer Pyrolytic Carbon Layer

The OPyC coating layer is the final diffusion barrier for fission products and provides mechanical protection for the SiC layer during particle handling and during fuel form compaction operations. Irradiation-induced shrinkage of the OPyC leads to compression of the SiC layer because of its net shrinkage under fast-neutron irradiation during the fuel lifetime in the reactor core. This reduces the tensile stress in that layer. The OPyC serves as a redundant barrier to gaseous fission-product release. The anisotropy of the OPyC layer is limited to control dimensional changes during irradiation. The OPyC thickness is typically 35 to 40 µm.

4.2.6 Coated Particle

When the AGR program began, since a firm HTGR design had not yet been developed, the program decided to adopt the LEU fissile particle of the MHTGR and Gas Turbine Modular Helium Reactor (GT-MHR) designs as the reference fuel form for AGR-1: a 350-µm 19.7%-enriched UCO kernel. As design activities began under NGNP, evaluations were conducted to determine whether a single particle design instead of the fissile-fertile system could be used in the HTGR. While a complete assessment would have required significantly more design development, the initial study showed promise and thus a single particle was adopted for AGR-2: a 425-µm 14.0%-enriched UCO kernel.

In terms of fuel particle design, a relationship exists among the actual values of the kernel size, buffer volume, and the maximum burnup to achieve consistent fuel performance. The physical size of the particle components is up to the designer and the achievable burnup depends on the particle enrichment and core design. One such metric of fuel performance is the tensile stress in the SiC layer, which depends on the pressure of fission gas in the buffer. Assuming 100% fission gas release and the maximum burnup, the stress in the SiC layer is proportional to the following attributes:

$$\sigma \propto \frac{B * V_k}{V_b} * \frac{r_{SiC}}{t_{SiC}} \qquad \textbf{Equation 4-1}$$

where:

σ	=	Tensile stress
B	=	Maximum burnup
V_k	=	Volume of kernel
V_b	=	Volume of buffer
r_{SiC}	=	Radius of SiC layer
t_{SiC}	=	Thickness of SiC layer

Table 4-2 shows these physical attributes for a variety of particles that have been proposed or used in HTGRs. This includes previous U.S. LEU fissile/fertile dual particles designs, an HEU kernel for the DOE NPR project, the historical German TRISO particle, and the Japan Atomic Energy Research Institute (JAERI) TRISO particle used in their HTTR and the value proposed for their advanced commercial design. Normalizing all the tensile stress metrics to the historic German value shows the metrics are within 20% of the German value indicating the tensile stress component should be similar in all these particle designs. (This small deviation is because of rounding the physical size of the kernel and buffer for ease of fabrication.) It should be noted this analysis does not credit the very important role irradiation-induced PyC shrinkage plays in developing a strong compressive component to the stress in the SiC layer.

The values of thicknesses of the PyC and SiC layers are based on the successful German program and are used by the Chinese as well. The Japanese use slightly different dimensions. The AGR program adopted the German coating thicknesses for its fuel development.

Table 4-2
Particle design attributes contributing to tensile stress in SiC

Parameter	German	JAERI HTTR	JAERI Advanced	U.S. LEU Fissile	U.S. Fertile	U.S. NPR	AGR
	Particle Design Parameters						
Kernel Composition	UO_2	UO_2	UO_2	UCO	UCO	UCO	UCO
Kernel Diameter (μm)	500	600	550	350	500	200	425
Buffer Thickness (μm)	95	60	100	100	65	100	100
IPyC Thickness (μm)	40	30	35	35	40	50	40
SiC Thickness (μm)	35	30	35	35	35	35	35
OPyC Thickness (μm)	40	45	40	40	40	40	40
Enrichment (%)	10.6	6	10	19.9	0.7	93	14.0
Burnup (% FIMA)	10	3.6	10	26	6	80	17
	Calculated Values						
Particle Diameter (μm)	920	930	970	770	860	650	855
Kernel volume (mm^3)	0.065	0.113	0.087	0.022	0.065	0.004	0.040
Buffer volume (mm^3)	0.107	0.082	0.134	0.065	0.065	0.029	0.088
Simple tensile stress metric	0.676	0.643	0.763	0.799	0.608	0.816	0.785
Normalized to German value	1.00	0.95	1.13	1.18	0.90	1.21	1.16

To understand the behavior of the coating layers as a coating system requires more detailed modeling. The basic behavior of the three coating layers of the TRISO-coated particle is shown in Figure 4-2. Fission gas pressure builds up in the kernel and buffer regions, while the IPyC, SiC, and OPyC act to retain this pressure. The IPyC and OPyC layers both shrink and creep during irradiation of the particle while the SiC exhibits only elastic response. A portion of the gas pressure is transmitted through the IPyC layer to the SiC. This pressure increases as irradiation of the particle progresses, thereby contributing to a tensile hoop stress in the SiC layer. Countering the effect of the pressure load is the shrinkage of the IPyC and OPyC layers during irradiation, which causes them to push or pull inward on the SiC. Due to anisotropy in the PyC shrinkage behavior, the shrinkage histories differ in the radial and tangential directions. The shrinkage in the radial direction reverses to swelling at moderate fluence levels, whereas shrinkage in the tangential direction continues to high fluence levels.

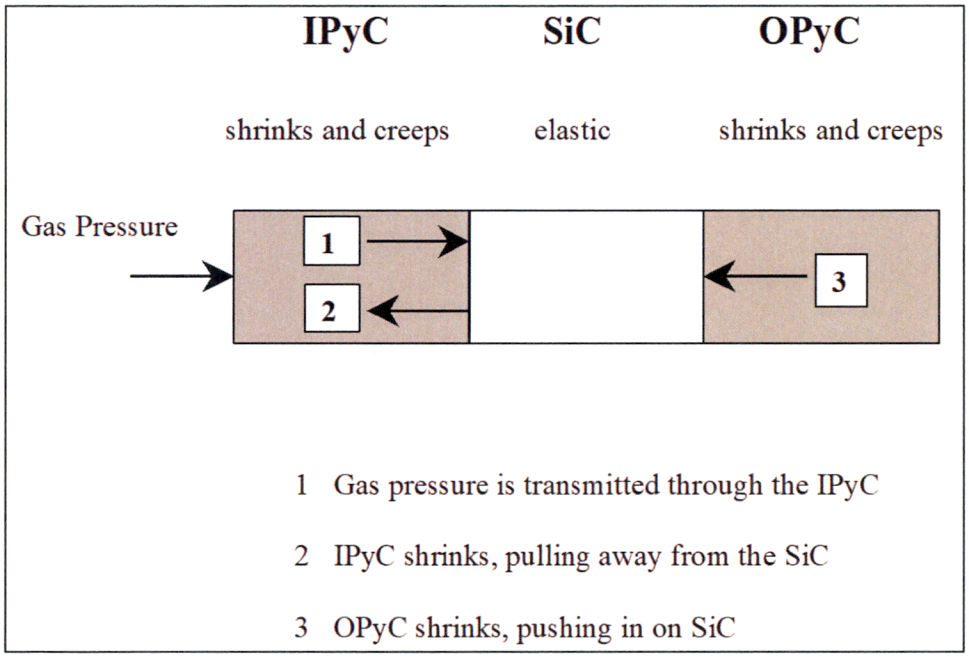

Figure 4-2
Behavior of coating layers in a fuel particle
Courtesy of Idaho National Laboratory and used with permission of Battelle Energy Alliance, LLC

In the stress analyses of most models, an internal pressure is applied to the IPyC to simulate the fission gas build-up. The shrinkage strain rates and creep coefficients for the PyC and the elastic properties for the PyC and the SiC are based from data compiled in a report by GA in 1993 [28]. As such, the shrinkage strains are treated as functions of four variables: fluence level, pyrocarbon density, degree of anisotropy as measured by the Bacon Anisotropy Factor (BAF), and irradiation temperature. Irradiation-induced creep is treated as secondary creep, with a coefficient that is a function of PyC density and irradiation temperature. The creep coefficients used in the analyses described herein were set equal to twice the values recommended in the GA data. This is closer to what is used in older performance models [29-31] and has resulted

in predictions that are in better agreement with results from irradiation experiments of the NPR – MHTGR Program [32]. The elastic modulus for the PyC layers is applied as a function of four variables (the same variables as used for shrinkage), while the elastic modulus for the SiC is applied as a function of temperature only.

Figure 4-3 plots a time evolution for the tangential stress at the inner surface of the SiC layer for a normal spherical particle which is irradiated to a fluence level of 3.0×10^{25} n/m^2. Early during irradiation, the shrinkage of the PyC layers induces an increasing compressive stress in the SiC. Eventually, creep in the PyC layers relieves stress in those layers, diminishing the beneficial effect of the shrinkage. Therefore, the tangential stress in the SiC reaches a minimum value, and then steadily increases through the remainder of irradiation. A pressure vessel failure is expected to occur if the tangential stress reaches a tensile value that exceeds the strength of the SiC for that particle.

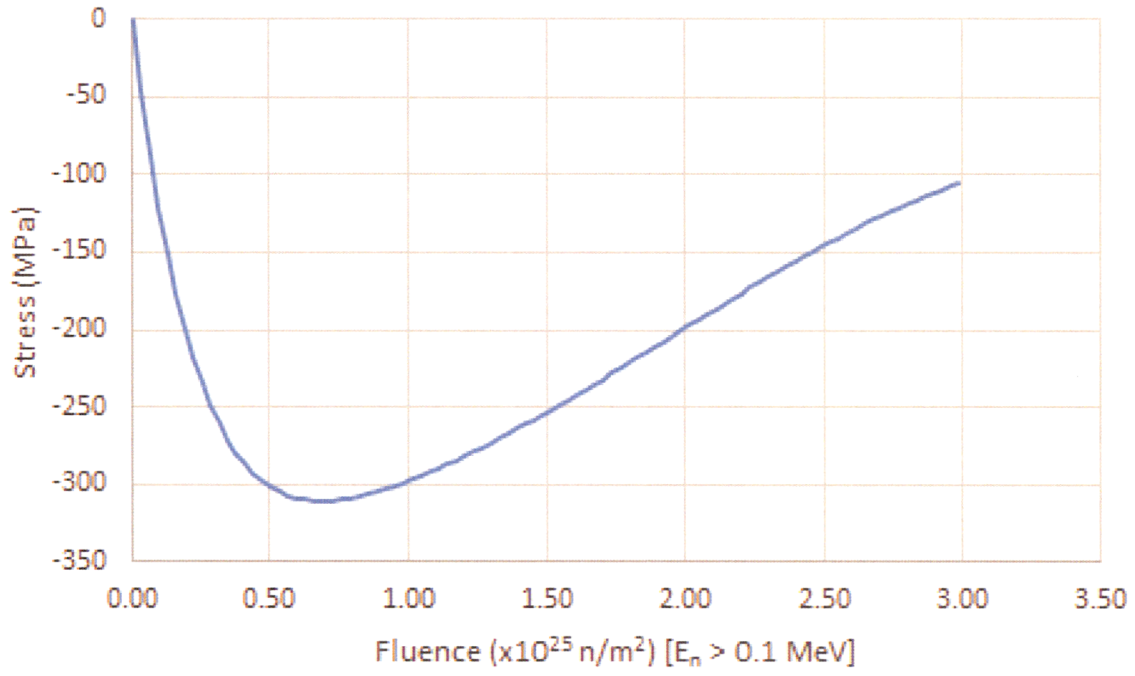

Figure 4-3
Tangential SiC stress history for a normal particle
Courtesy of Idaho National Laboratory and used with permission of Battelle Energy Alliance, LLC

Sensitivity studies have been conducted using a thermomechanical fuel performance model to understand other potential failure modes of the particles (for example, cracking of the IPyC layer, excessive asphericity [33], thinning of the SiC layer [34]). In addition, the model has been used to study the impact of fuel particle attributes on the calculated stresses in the particles. Many of the coating properties measured during the AGR program evolved from those developed by GA based on their historical experience at FSV, modified as necessary to assure the high-level radionuclide release criteria could be met for their HTGR designs. During the

AGR program, more systematic calculations have been performed to determine which of the measured fuel attributes are the most critical from a fuel performance perspective and what are the appropriate critical limits for those attributes to be used in a specification. Results of the PARFUME analysis [35] indicate:

- Many of the fuel attributes have minimal impact on the thermomechanical performance of TRISO-coated UCO particles. The nominal thicknesses and densities of the German coatings are adequate for high-temperature reactor applications, and in many cases, there is performance margin. However, given the large experimental basis for these coatings, the models were not used to optimize/change layer thicknesses or densities from the German values.
- Minimal change was observed in the overall TRISO-coated particle failure probability as the PyC density (both IPyC and OPyC) and anisotropy were varied over the typical range of values. This is probably due to the uncertainties in the material properties, especially irradiation-induced creep.
- When varying the thickness of the SiC layer, the failure probability increased as the thickness decreased because there is less structural material to retain the fission gas pressure and subsequent increase in tangential stress in the layer. Thus, a critical limit[12] on the minimum thickness of SiC is warranted.
- Conversely, failure probability increased as the IPyC layer thickness increased because thicker PyC experience higher stress levels early in irradiation. This results in a higher IPyC cracking probability causing localized stress concentrations in the SiC layer. Thus, a critical limit on the maximum IPyC thickness is warranted.
- As the buffer thickness decreases, the volume available to store fission gas decreases resulting in a higher pressure and higher stress in the SiC layer. Thus, a critical limit on the minimum buffer thickness is warranted.
- For aspherical particles, as characterized by the aspect ratio (that is, largest diameter divided by smallest diameter on a particle), the model used in the analysis treats asphericity essentially as a flat plate on one side of the particle. Increasing the aspect ratio increases the surface area of the flat plate increasing the stress in the SiC layer due to pressure accumulation. Thus, a critical limit on aspect ratio is warranted.

These critical limits have been incorporated into the AGR fuel specification.

4.3 Failure Mechanisms

The following failure mechanisms have been identified as capable of causing partial or total failure of the TRISO-coating system under irradiation and/or during postulated accidents:

- Pressure vessel failure of standard ("intact") particles (particles without manufacturing defects)
- Pressure vessel failure of particles with defective or missing coatings
- Irradiation-induced failure of the OPyC coating

[12] A critical limit specifies that less than 1% of the population can have values above (upper critical limit) or below (lower critical limit) that limit at 95% confidence depending on the attribute in question.

- Irradiation-induced failure of the IPyC coating and potential SiC cracking
- Failure of the SiC coating caused by kernel migration in the presence of a temperature gradient
- Failure of the SiC coating caused by fission-product/SiC interactions
- Failure of the SiC coating caused by CO/SiC interactions
- Failure of the SiC coating resulting from thermal decomposition
- Failure of the SiC coating caused by heavy metal (HM) dispersion in the buffer and IPyC coating layers

These mechanisms are shown schematically in Figure 4-4. Phenomenological performance models, typically inspired by first principles and correlated with experimental data, have been developed to model each of these mechanisms [36,37].

As-manufactured HM contamination is not an in-service failure mechanism, but is very important with respect to fission product release. It is an extreme case of as-manufactured coating defects, whereby trace amounts of HM are not encapsulated by a single intact coating layer (analogous to "tramp uranium" in LWR fuel). Modern fuel product specifications allow only very small fractions of HM contamination (~10^{-5} is typical). Nevertheless, when exposed kernel defects and in-service failure fractions are low, HM contamination can become a dominant source of fission product release.

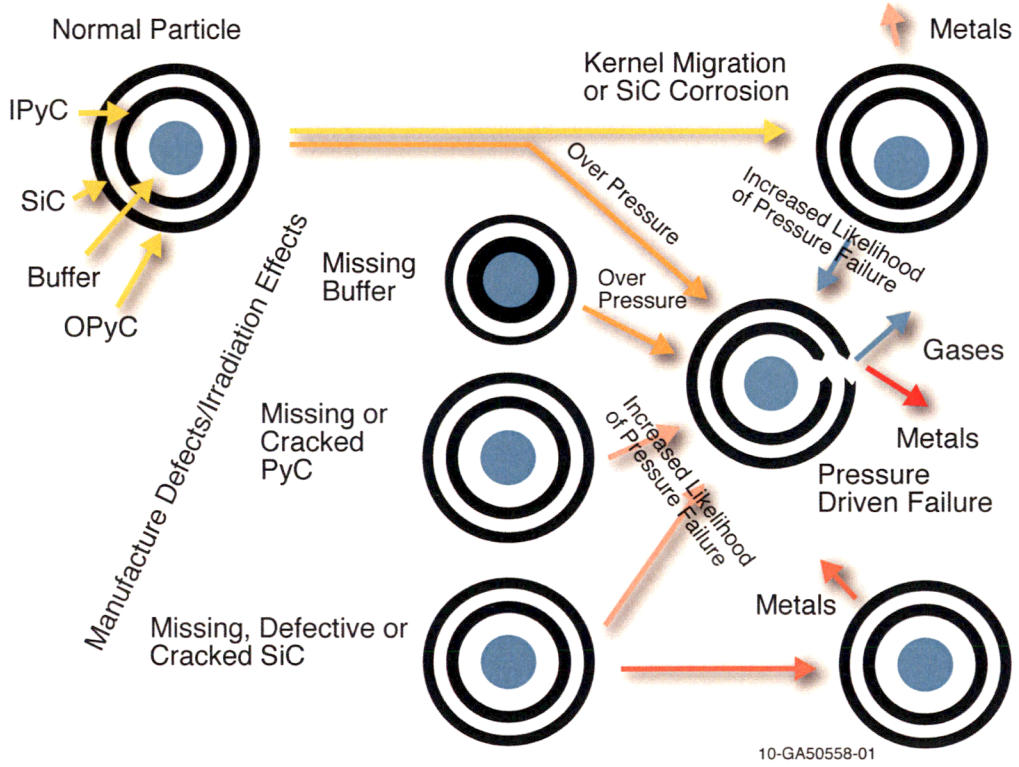

Figure 4-4
TRISO particle failure mechanisms
Courtesy of Idaho National Laboratory and used with permission of Battelle Energy Alliance, LLC

The observed failure mechanisms for TRISO fuel can be categorized as structural/mechanical or thermochemical in nature. Failure mechanisms in both categories can be affected by the release of excess oxygen during fission and subsequent formation of carbon monoxide. The various failure mechanisms are discussed in greater detail in the following sections.

4.3.1 Structural/Mechanical Mechanisms

During irradiation, long-lived and stable fission gases are released from the kernel into the buffer, which increases the internal gas pressure. For some particle designs (for example, UO_2 TRISO), carbon monoxide can also be generated during irradiation, which further increases the gas pressure. Because the SiC layer has a much higher elastic modulus than the PyC layers,[13] it would bear most of the internal pressure force, which produces a tensile stress if the irradiation-induced dimensional changes of the PyC and SiC were comparable. However, the PyC layers shrink during irradiation, subjecting the SiC layer to compression. Within the range of allowed fuel service conditions (for example, temperature and fast neutron fluence), the compressive forces from PyC shrinkage more than compensate for the tensile stresses from internal pressure, such that the SiC remains in compression provided at least one of the PyC layers remains intact. From a structural/mechanical perspective, the SiC layer will remain intact provided it remains in compression or the tensile stress in the SiC layer does not exceed its strength.

4.3.1.1 PyC Performance

As discussed above, shrinkage of the PyC layers during irradiation is a favorable attribute, as the resulting compressive forces acting upon the SiC layer counteract the tension arising from fission gas pressure. PyC shrinkage produces tensile stresses in the PyC layers themselves, which can lead to failure of these layers. The strains and stresses generated in the PyC layers are complex functions of fast neutron fluence, irradiation temperature, and coating-material properties.

A property to which PyC performance is quite sensitive is *anisotropy*, which can be quantified using x-ray or optical diffraction techniques. Anisotropy is usually expressed in terms of the BAF. For a perfectly isotropic material, BAF = 1, and for a perfectly oriented medium, BAF = ∞. Sufficiently isotropic PyC layers (BAF $\leq$1.035) are able to perform well out to high fast neutron fluences because the irradiation-induced strains and stresses are relaxed to some extent by irradiation-induced creep.

4.3.1.2 Irradiation Induced Failure of IPyC Leading to SiC Cracking

PIE of fuel from the HRB-21 irradiation and the NP-MHTGR irradiations coupled with mechanical analyses showed fuel particle failures in these irradiation experiments were caused by irradiation-induced failure (cracking) of anisotropic IPyC. This leads to increased tension in the adjacent SiC layer to which it is bonded, increasing the probability of cracking the SiC layer [16,17]. These failure analyses led to changes in the coating conditions used in the fabrication of fuel particles in the AGR program [38] to ensure IPyC coatings with sufficient isotropy were produced.

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[13] In other words, SiC is much stiffer than PyC. Because of this property, it is reasonable to assume the IPyC and OPyC are isolated from each other when evaluating performance of these layers and overall performance of the TRISO-coating system.

4.3.1.3 Pressure Vessel Failure

In the absence of compressive forces from the PyC layers, the tensile stress, σ_{SiC}, in the SiC layer may be calculated with reasonable accuracy using the thin-shell approximation:

$$\sigma_{SiC} = \frac{Pr_{SiC}}{2t_{SiC}} \qquad \textbf{Equation 4-2}$$

where

P = Internal pressure inside the particle

r_{SiC} = Radius to the middle of the SiC layer

t_{SiC} = Thickness of the SiC layer

Pressure vessel failure occurs when the tensile stress in the SiC layer exceeds the strength of the SiC layer. The fraction of particles with a failed SiC coating,[14] f_{SiC}, is calculated using Weibull statistical strength theory, assuming volume flaws and a uniform stress distribution in the SiC layer, as:

$$f_{SiC} = 1 - \exp\left[-\left(\frac{\sigma_{SiC}}{\sigma_o}\right)^m V_{SiC}\right] \qquad \textbf{Equation 4-3}$$

Where:

σ_o ≡ Weibull characteristic strength

m ≡ Weibull modulus

V_{SiC} ≡ Volume of the SiC layer.

4.3.2 Thermochemical Mechanisms

Fuel failure caused by thermochemical mechanisms can be limited in large measure through the nuclear and thermal-hydraulic design of the reactor core. For the fuel to satisfy performance criteria, peak fuel temperatures must be kept sufficiently low, and the fraction of fuel that experiences relatively high temperatures for long periods of time must be kept sufficiently small. Thermochemical failure mechanisms that have been observed to occur in coated-particle fuel are described below.

4.3.2.1 Kernel Migration

Local fuel temperatures and temperature gradients across the fuel can be relatively high when the reactor is producing power. Under these conditions, oxide and carbide fuel kernels can migrate up the thermal gradient. This phenomenon is often referred to as the "amoeba effect" and can lead to complete failure of the coating system. For oxide kernels, migration may be caused by carbon diffusion or gas-phase diffusion of CO or other gaseous carbon compounds [39]. Failure by this mechanism is correlated as a function of temperature, thermal gradient, and thicknesses of the buffer and IPyC layers. Failure is assumed to occur when the kernel material contacts the

[14] This fraction applies to the population of particles that have a failed IPyC layer and a failed OPyC layer.

SiC layer. The particle-to-particle variations in the buffer and IPyC thicknesses (expressed as normal distributions with measured variances) are accounted for when calculating the failure probability. In UCO kernels, this failure mechanism is not observed because CO or other gaseous carbon compounds are greatly reduced.

4.3.2.2 Chemical Attack of SiC

Noble metals (for example, Ru, Rh, Pd, and Ag) are produced during fission of uranium, in some cases with relatively high yield. During irradiation, the thermochemical conditions are not conducive to the formation of stable oxides of these elements. They readily migrate out of the fuel kernel, regardless of its composition. Reactions of SiC with Pd have been observed during PIE of TRISO fuel [40]. Although the quantity of Pd is small compared with the mass of the SiC layer, the reaction is highly localized, and complete penetration of the SiC layer can occur if high temperatures are maintained for a sufficient period of time (see Figure 4-5). The reaction rate is highly dependent on temperature. The time required to penetrate the SiC layer decreases rapidly as the temperature increases above about 1300°C.

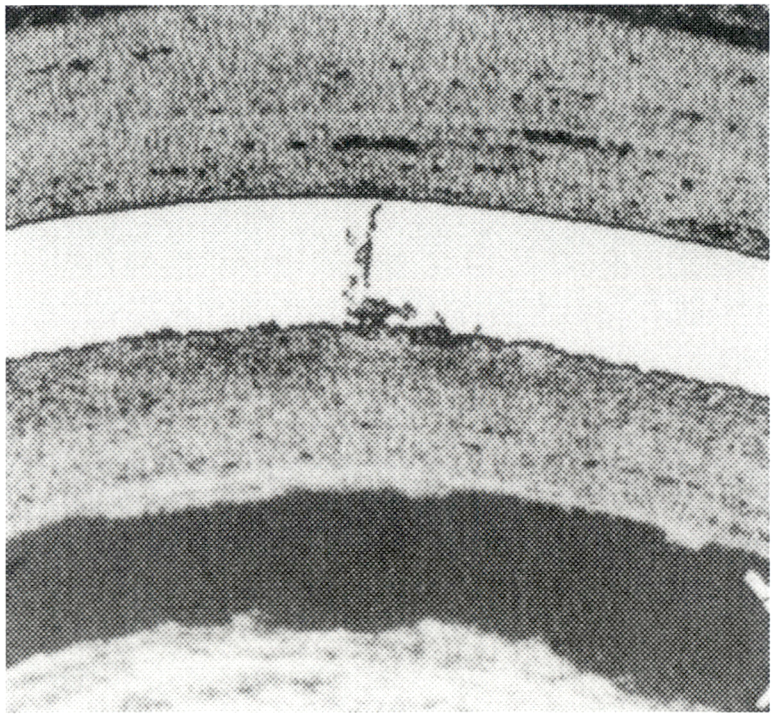

Figure 4-5
Localized fission-product attack of the SiC layer in an irradiated UCO particle from the HRB-16 experiment
Courtesy of Idaho National Laboratory and used with permission of Battelle Energy Alliance, LLC

Excess oxygen is produced in a UO_2 kernel during irradiation because the oxygen liberated by fission is not completely consumed by reactions with fission products. At low burnup, some of the excess oxygen may remain trapped in the kernel. At high burnup the kernel becomes more porous, and it is likely nearly all of the oxygen will escape a UO_2 kernel, after which it will quickly react with carbon in the buffer to predominately form CO. Excessive CO not only increases the pressure vessel and kernel migration failure probabilities, but can also corrode the SiC layer at accident-condition temperatures.

Chemical attack of the SiC layer by CO has been observed in UO_2 particles irradiated at temperatures above approximately 1400°C [41]. Degradation occurred near locations where the IPyC layer was cracked. The kernels of particles with degraded SiC layers were examined with an electron microprobe, which showed the presence of silicon in the form of fission product silicides. Thermo-chemical calculations supported the hypothesis that silicon is transported to the kernel in the form of silicon monoxide (SiO) gas produced by the reaction of CO with SiC. The SiO subsequently reacts with fission products.

4.3.2.3 Thermal Decomposition of the SiC Layer

At very high temperatures, SiC will decompose into its constituent elements. The silicon vaporizes, leaving a porous carbon structure. Based on calculations performed for previous core designs, this failure mechanism is not an important contributor to fuel failure at normal operating or postulated accident conditions. However, thermal decomposition of SiC occurs rapidly at temperatures above 2000°C.

4.3.2.4 Relationship between Fuel-Failure Mechanisms and Fuel-Particle Properties

The fuel service conditions and parameters that influence the fuel failure mechanisms are summarized in Table 4-3 below. The fuel particles must be designed and manufactured such that the properties defined in Section 5.3 are within limits that result in acceptable fuel performance (for example, fission product retention). The failure mechanisms are correlated with the reactor service conditions in models that are used to predict fuel performance. In addition, sensitivity studies have been conducted to assess the relative impact of various properties on calculated failure fractions, and results are summarized in Section 4.2.

4.4 Performance Bases

At the start of the AGR program, without a reactor design concept selected, the program decided to qualify fuel to an operating envelope that would bound potential options across a range of high-temperature reactor conceptual designs. Figure 4-6 is a radar plot of the five most important parameters for qualifying fuel performance. The parameters are listed below along with an explanation of their importance in influencing fuel performance:

- **Fuel temperature**. Many of the potential failure mechanisms and fission product transport mechanisms are dependent on both time at temperature during power operation and time at temperature under postulated accident condition.
- **Fuel burnup**. Determines the quantity of fission products in the kernel and thus the gas pressure and fission product concentration in the particles that can interact with the coating layers.

- **Fuel fast fluence**. Determines the level of radiation damage in the particles and the potential changes in properties and dimensions in the layers.
- **Power density**. Together with the thermal conductivity and the geometry of the fuel (for example, compact, pebble) determines the temperature gradient across the fuel specimen as some potential failure mechanisms depend on this temperature gradient. Note that the power density in Figure 4-6 is for the entire core volume not just the fuel specimen.
- **Particle packing fraction**. Packing fraction together with the global power density can be used to establish the power per particle, which establishes the temperature inside the particle.

Envelopes are shown in Figure 4-6 for the successful German and Japanese TRISO-coated particle fuel programs established in the 1980s and 1990s, respectively, along with bounds anticipated for the NGNP designs. The AGR program used the NGNP envelope to guide its irradiation testing.

Table 4-3
Relationship between mechanisms of fuel failure properties of fuel particles

Failure Mechanism	Service Conditions	Parameters Influencing Failure Mechanism
Pressure vessel failure	Temperature, burnup, fast fluence	Strength of SiC Buffer density (void volume) Fission-gas release Kernel type (CO production) Layer thicknesses IPyC and OPyC performance
Irradiation-induced PyC failure leading to SiC cracking	Fast fluence, temperature	Dimensional change of PyC Irradiation-induced creep of PyC Anisotropy of PyC Strength of PyC PyC thickness PyC density Tensile stress in SiC at IPyC crack tip SiC strength
IPyC partial debonding	Temperature, fast fluence	Nature of IPyC-SiC interface Interfacial strength Dimensional change of IPyC Irradiation-induced creep of IPyC
Kernel migration	Temperature, burnup, temperature gradient	Kernel type (UO_2, UCO, and so on) Buffer and IPyC thickness
Diffusive release through intact layers	Temperature, burnup, temperature gradient, time at temperature	Chemical state/transport behavior of fission products Microstructure of SiC SiC thickness
Fission product attack of SiC	Temperature, burnup, temperature gradient, time at temperature	Chemical state/transport behavior of fission products Kernel type (UO_2, UCO, and so on) Microstructure of IPyC and SiC
Corrosion of SiC by CO	Temperature, burnup, time at temperature	Kernel type (UO_2, UCO, and so on) IPyC integrity
SiC thermal decomposition	Temperature, time at temperature	SiC thickness SiC microstructure
SiC permeability/SiC degradation	Burnup, temperature, fast fluence	Microstructure of SiC Thickness of SiC Permeability of SiC SiC layer impurities from fabrication process

This envelope resulted in the need for a fuel form that could survive at peak fuel temperatures of 1250°C on a time-average basis and high burnups in the range of 150 to 200 GWd/MTHM (metric tons of heavy metal) or 16.4 to 21.8% FIMA. The program selected UCO as the fissile kernel of choice because of its ability to limit CO production and kernel migration under irradiation phenomena that in 2003 were considered life limiting in the traditional UO_2 TRISO fuels if they were to operate at the upper temperature range (~1250°C) and high burnup anticipated in some of the designs.

For comparison, and as discussed in detail in Section 6, the UCO TRISO fuel compacts in the AGR-1 irradiation had a packing fraction of 37% and achieved burnups of between 11.3 and 19.6% FIMA and fast fluences between 2.2 and 4.3 × 10^{25} n/m^2 (E >0.18 MeV). Peak time-average temperatures ranged from 1069 to 1197°C and time-average volume-average (TAVA) temperatures ranged 955 to 1136°C. The UCO TRISO fuel compacts in the AGR-2 irradiation also had a packing fraction of 37% and achieved burnups between 7.3 and 13.2% FIMA and fast fluences between 1.9 and 3.5 × 10^{25} n/m^2 (E > 0.18 MeV). Peak time-average temperatures ranged from 1080 to 1360°C and TAVA temperatures ranged from 987 to 1296°C. In both AGR-1 and AGR-2, capsule average power densities ranged from 4 to 14 W/cm^3 and power per particle ranged between 20 and 160 mW/particle during the irradiations.

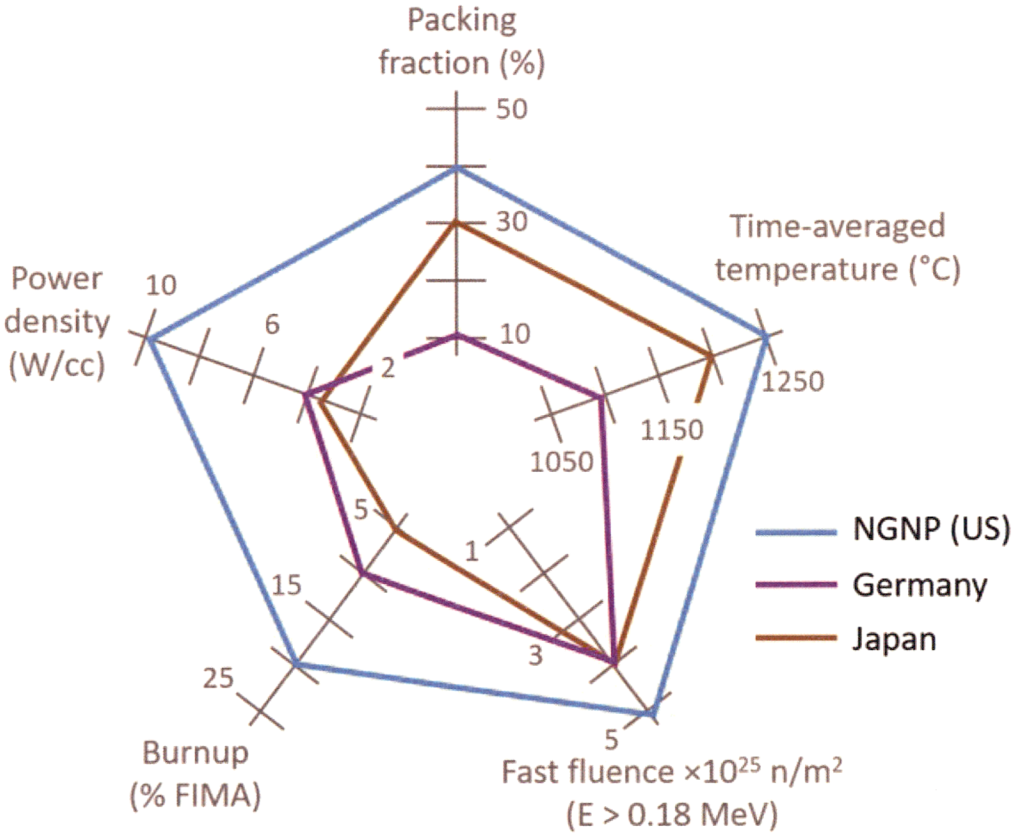

Figure 4-6
Radar plot of key parameters for TRISO-coated fuel performance. Germany and Japan plots represent historic values; NGNP indicates the performance envelope anticipated by the U.S. fuel development program.
Courtesy of Idaho National Laboratory and used with permission of Battelle Energy Alliance, LLC

5
ADVANCED GAS REACTOR FUEL DEVELOPMENT AND QUALIFICATION PROGRAM

5.1 Program Background and Objectives

In fiscal year 2002, the DOE Office of Nuclear Energy, Science, and Technology initiated development of the Advanced Gas Reactor Fuel Development and Qualification Program for coated-particle fuel. The resulting *Technical Program Plan for Advanced Gas Reactor Fuel Development and Qualification Program* [39] and subsequent revisions defined fuel development activities to support licensing and operating an HTGR in the U.S. under the umbrella of the NGNP project in accordance with the Energy Policy Act of 2005 when it was enacted. The AGR program is currently part of the DOE Advanced Reactor Technologies (ART) program and is pursuing irradiation and performance data for the qualification of TRISO particle fuel for use in high-temperature reactors [38].

TRISO particle fuel development and qualification activities support multiple HTR designs, including prismatic and pebble-bed HTGRs as well as FHRs. The AGR program to date has focused on manufacturing and testing the fuel design for HTR concepts using the most recent GT-MHR fuel product specification as a starting point [1]. Irradiation, safety testing, and PIE plans support fuel development and qualification in an integrated manner. Preliminary operating conditions and performance requirements for the fuel and preliminary fuel product specifications to guide the AGR program's fuel fabrication process development activities were based on previously completed HTGR design and technology development activities, operating conditions, and performance requirements.

A complete set of fuel design specifications for an HTGR has not been developed which could be used in the AGR program, but the maximum burnup envisioned in a prismatic HTGR is within the range of 150 to 200 GWd/MTHM or 16.4 to 21.8% FIMA. Maximum burnups for pebble-bed designs have historically been considerably less than this. Although Germany has demonstrated excellent performance of UO_2 TRISO particle fuel up to about 10% FIMA and 1150°C, UO_2 fuel is known to have limitations because of CO formation, including kernel migration at the higher burnups, power densities, temperatures, and temperature gradients that may be encountered in the prismatic HTGR design, and CO corrosion of the SiC layer. With UCO fuel, the kernel composition is engineered to minimize CO formation and kernel migration, which are key threats to fuel integrity at higher burnups, temperatures, and temperature gradients. The performance of German SiC-based, TRISO-coated-particle, UCO fuel up to 22% FIMA (as measured by the in-pile gas release in irradiation test FRJ2-P24 [18]) and the excellent performance of U.S.-made UCO TRISO fuel in AGR-1 and AGR-2 give added confidence that high-quality SiC-based, TRISO-coated-particle, UCO fuel can be made and its superior irradiation performance demonstrated statistically.

In addition to excellent fission product retention during normal operation at high burnups and high temperatures, HTGR fuel must exhibit satisfactory fission product retention under postulated accident conditions. Limited data on the accident performance of SiC-based TRISO-coated UO_2 fuel at high burnups indicate increased cesium releases at burnups ≥14% FIMA, so safety testing is an important element of any fuel qualification effort. The AGR program chose to develop coated-particle fuel using a low-enriched UCO kernel to qualify a fuel to meet fuel performance requirements under specified fuel service conditions. Thus, SiC-based TRISO-coated UCO was chosen as the baseline AGR fuel to be fabricated and tested. This fuel development path complemented particle fuel development with a UO_2 kernel that was being pursued by South Africa, China, and Europe. Safety testing of irradiated AGR-1 and AGR-2 UCO TRISO compacts has demonstrated the fuel's robust behavior for about 300 hours at 1600, 1700, and 1800°C, giving added confidence that SiC-based TRISO particle fuel can meet safety performance requirements (see Sections 6 and 7).

The TRISO-coated UCO fuel specification [42] utilizing SiC as the primary fission product retention layer was developed in response to extensive evaluations [16,17] of the fuel failures experienced in irradiations in the NP-MHTGR and the MHTGR programs. This was the starting point for the fuel specification developed for the current program [43]. It was expected this fuel would exhibit acceptable fuel performance at higher burnups (16 to 22% FIMA) time-average fuel temperatures up to 1250°C for normal operation and 1600°C for potential accident conditions, and fast neutron fluences up to 5×10^{25} neutrons/m^2.

The AGR program was established to achieve the following overall goals:

- Provide a fuel qualification data set in support of the licensing and operation of an HTGR. HTGR fuel performance demonstration and qualification comprise the longest duration research and development (R&D) tasks required for design and licensing. The fuel form is to be demonstrated and qualified for service conditions, which include normal operation and potential accident scenarios.
- Support deployment of HTGRs for hydrogen, process heat, and energy production in the U.S. by reducing market entry risks posed by technical uncertainties associated with fuel production and qualification.
- Extend the value of DOE Office of Nuclear Energy resources by using international collaboration mechanisms where practical.
- Establish a domestic TRISO particle fuel manufacturing capability for fabricating demonstration and qualification experiment fuel.
- Improve understanding of the fabrication process, its impact on as-fabricated fuel properties and attributes, and their impacts on in-reactor performance.

At the onset of the AGR program in 2002, facilities and personnel experienced in activities necessary to address the program goals existed in the U.S., primarily at INL and Oak Ridge National Laboratory (ORNL). INL and ORNL personnel with experience and knowledge of TRISO particle fuel, facility status, and capabilities were involved in developing the initial *Technical Program Plan for the Advanced Gas Reactor Fuel Development and Qualification Program* [44]. In addition, GA provided input regarding prismatic HTGR fuel performance requirements and perspectives from its experience in fuel development, fuel fabrication, and

fuel-related analytical capabilities needed to support licensing interactions. BWX Technologies Inc. (BWXT) provided input based on its experience and capabilities for fuel-kernel production and fuel-particle coating. Many of the individuals who helped develop this plan were directly involved in producing and testing previous U.S. fuel for the MHTGR and the NP-MHTGR. They conducted extensive investigations and reviews in the early 1990s following the unexpectedly high fuel failure levels observed in those tests.

Following review by the NGNP project by the NEAC [45], DOE halted design-specific efforts on the NGNP project at the end of the conceptual design phase, in part because a viable public-private partnership for a demonstration reactor and follow-on commercialization was not established. To date no partnership has been formed, although recently several private companies have expressed interest in using UCO TRISO fuel based on the AGR program design in an advanced HTR design. Thus, the AGR program focus is to qualify a fuel form and establish a commercial fuel vendor in the U.S. The HTGR R&D will not perform verification or validation of any potential reactor vendor codes.

The AGR program involves the following five major program elements:

1. **Fuel Fabrication**. This program element—to fabricate TRISO particle fuel (that is, manufacturing fuel that meets the fuel quality and performance requirements for licensing an HTR)—requires development of a coating process that replicates, to the greatest extent practical, the HTGR particle design and properties of the coatings on German fuel particles that have previously exhibited superior irradiation and accident performance. Coating-process development has been accomplished in two phases: initially in a 2-in.-diameter, laboratory-scale coater (AGR-1) followed by scale-up to a 6-in., prototypic, engineering-scale coater (AGR-2). The Fuel Fabrication program element has included establishing the fuel fabrication infrastructure; developing the process for the low-enriched uranium oxycarbide kernels, TRISO particles, and compacts; developing coating process models; developing quality control methods; performing fuel process scale-up analyses; and developing process documentation for technology transfer to private industry. The fuel fabrication effort has produced TRISO particle fuel within cylindrical fuel compacts that met fuel product specifications and provided fuel and material samples for characterization, irradiation, safety testing, and PIE as necessary to meet the overall AGR program goals.
2. **Fuel and Material Irradiation**. This program element provides data on fuel performance during irradiation to support fuel process development, qualify a fuel design and fabrication process for normal operating conditions, and support development and validation of fuel performance and fission product transport models and codes. This program element also provides irradiated fuel and materials necessary for PIE and safety testing. Seven irradiation tests, designated as AGR-1 through AGR-7, have been defined to provide data and sample materials within the AGR program.
3. **Fuel PIE and Safety Testing**. This program element provides the facilities and processes to measure the performance of TRISO particle fuel under normal operating and potential accident conditions. Moisture and air ingress testing in quantities expected to exist within the typical helium and neon gas supplies used during irradiation (testing performed during AGR-3/4 irradiation) and safety testing (planned to be performed during AGR-5/6/7 PIE) will be performed to determine their effects on TRISO particle fuel. This work supports the fuel manufacturing effort by providing feedback on the performance of kernels, coatings, and

compacts during irradiation and under potential accident conditions. PIE and safety testing provide a broad range of data on fuel performance and fission product transport within TRISO-coated fuel particles, compacts, and carbonaceous matrix materials representative of fuel element blocks. These data, in combination with the in-reactor measurements (irradiation conditions and fission gas release-rate-to-birth-rate ratios), are necessary to demonstrate compliance with fuel performance requirements and support developing and validating computer codes.

4. **Fuel Performance Modeling**. This program element addresses the structural, thermal, and chemical processes that can lead to TRISO-coated particle failures. It considers the effects of fission product chemical interactions with the coatings, which can lead to degradation of the coated-particle properties. Fission product release from the fuel particles and transport in the fuel compact matrix and fuel element carbonaceous matrix during irradiation are also modeled. Computer codes and models will be further developed and refined as appropriate in response to irradiation, PIE, and safety testing data.
5. **Fission Product Transport and Source Term**. This program element addresses the transport within reactor core materials of fission products produced in the TRISO particle fuel and is intended to provide a technical basis for source terms for HTGRs under normal irradiation and potential accident conditions. Most of this work scope has not been performed because of funding shortfalls and higher priority work scope. Some initial fission product transport studies were performed on hydrogen and tritium permeation through high nickel superalloys with results that were included in published reports. An evaluation of data from irradiation and safety testing of "designed-to-fail" fuel particles will be performed as part of the AGR-3/4 PIE, see description below. The purpose of the evaluation is to characterize fission product release and transport from TRISO particle fuel into fuel compact matrix and fuel element carbonaceous matrix under normal and off-normal HTGR conditions.

5.2 Overview of AGR Program Irradiations

The number and type of test trains to be irradiated were planned based on the needs of the fuel manufacturing, fuel performance modeling, and fission product transport activities. Seven experiments were identified based on discussions among the working groups during the course of developing the original program plan. Program budget constraints and further development of the test train designs have altered the type of test trains that were initially planned to be used for individual irradiations. In some cases, several originally planned individual experiments were combined into a single irradiation test train. This approach has taken advantage of the larger size of the ATR northeast flux trap (NEFT) irradiation position to accommodate a greater number of fuel specimens compared to the large B positions used for the AGR-1 and AGR-2 irradiations, allowing multiple experiment objectives to be accomplished in a single irradiation campaign. An eighth experiment, AGR-8, intended to provide radionuclide source term validation data was eliminated from the program plan in 2011 due to budget constraints and the absence of a reactor design effort going forward.

The four irradiation campaigns in the AGR program are outlined below.

5.2.1 Early Fuel Experiment (AGR-1)

This multi-monitored capsule test train included six capsules, each containing 12 compacts made from TRISO particles produced in a small laboratory-scale (2-in.) coater in conjunction with fuel process development. This irradiation experiment provided experience with a multi-monitored test train design, fabrication, and operation, which facilitated the design, fabrication, and operation of subsequent irradiation experiments. The AGR-1 irradiation provided data on irradiated fuel performance for baseline and fuel variants that were selected based on data from fuel process development and existing irradiation experience. The early data on the performance of fuel variants supported the selection of a reference fuel for the AGR-2 irradiation experiment and development of an improved fundamental understanding of the relationship among the fuel fabrication process, as-fabricated fuel properties, normal operation, and potential accident condition performance.

5.2.2 Performance Test Fuel Experiment (AGR-2)

This multi-monitored capsule test train included six independent capsules and featured a design very similar to AGR-1. Four of the capsules contained fuel manufactured in the U.S.: three capsules contained UCO fuel compacts and one capsule contained UO_2 fuel compacts. The U.S. UCO and UO_2 TRISO particles were fabricated in an engineering-scale 6-in. coater using process conditions derived from the production of AGR-1 Variant 3 (SiC layer produced using a mixture of hydrogen and argon diluent gases). Fuel compacts were fabricated using laboratory-scale processes and equipment at ORNL. The UCO compacts were subjected to a range of burnups and temperatures exceeding anticipated reactor service conditions in all three capsules. The two remaining capsules contained fuel manufactured by Westinghouse/Pebble-Bed Modular Reactor SOC Ltd., and Commissariat à l'Énergie Atomique et Aux Énergies Alternatives (the fabrication and performance of this fuel is not discussed in this report). This test train provided irradiated fuel performance data for coated particles fabricated at the engineering scale. It also provided fuel specimens for PIE and safety testing. The data obtained from the AGR-2 irradiation and subsequent PIE and safety testing further increase the fundamental understanding of the relationship among the fuel fabrication process, as-fabricated fuel properties, normal operation, and potential accident condition performance.

5.2.3 Fission Product Transport Experiments (AGR-3/4)

This multi-monitored capsule test train was a combination of the AGR-3 and AGR-4 experiments originally planned as separate irradiations in large B positions but were combined and placed in the NEFT. This test train included compacts containing TRISO-coated "driver" fuel particles as well as 20 "designed-to-fail" (DTF) fuel particles, each within rings of carbonaceous material. DTF fuel particles for use in fission product transport testing consisted of reference kernels with only a ~20-μm-thick pyrocarbon seal coating that was intended to fail during irradiation and provided known fission product source terms. The test train was designed to provide data on fission product diffusivities in fuel kernels and sorptivities and diffusivities in compact matrix and carbonaceous matrix materials for use in upgrading fission product transport models. The AGR-3/4 experiments also have provided irradiated fuel performance data on fission product gas release from failed particles and irradiated fuel samples for PIE. The in-pile

gas release and PIE data on fission gas and metal release from kernels will be used in developing improved fission product transport models to the extent possible from the experimental results. As this experiment was focused on fission product transport and not fuel performance, the results are not discussed in this report.

5.2.4 Fuel Qualification and Fuel Performance Margin Testing Experiments (AGR-5/6/7)

This multi-monitored capsule test train is a combination of the AGR-5, AGR-6, and AGR-7 experiments, which were planned originally for separate irradiations in large B positions, similar to AGR-1 and AGR-2, but were combined for irradiation in the NEFT. The test train includes a single fuel particle type, fabricated using process conditions and product parameters considered to provide the best prospects for successful performance based on process development results and available data[15] from AGR-1 and AGR-2 irradiations. This is the reference fuel design selected for qualification. Variations in capsule conditions (burnup, fast fluence, and temperature) were established in the irradiation test specifications.

The AGR-5/6 portion of this test train will provide irradiated fuel performance data and irradiated fuel samples for safety testing and PIE in a sufficient quantity to demonstrate compliance with statistical performance requirements under normal operating and potential accident conditions.

The AGR-7 portion of this test train includes the same fuel type as used in AGR-5/6 and occupies one of the five capsules. The irradiation will test fuel substantially beyond its operating temperature envelope, so some measurable level of fuel failure is expected to occur. This fuel performance margin test will provide irradiation data and irradiated fuel samples for PIE and post-irradiation heat-up testing in sufficient quantity to demonstrate the capability of the fuel to withstand conditions beyond AGR-5/6 normal operating conditions in support of plant design and licensing.

The experiment is notable for including a larger population of particles than previous irradiations (total particle count is approximately 570,000) and for extending the range of irradiation temperatures beyond the AGR-1 and AGR-2 experiments. The AGR-5/6 portion of the experiment is intended to contain particles with time-average irradiation temperatures ranging from 600 to 1400°C. The AGR-7 capsule contains a sub-population of approximately 54,000 particles and will have a time-average peak temperature of 1500 ±50°C.

The AGR-5/6/7 irradiation experiment began in February 2018 and is expected to operate for approximately 3 years. The ongoing irradiation will not be discussed in this report.

[15] The decision to proceed with fabrication of qualification test fuel was made based on information available at the time, which included full irradiation of AGR-1 plus PIE, heat-up and fission product metal release data on AGR-1 fuel, as well as in-pile gas release data from AGR-2.

5.3 Summary of AGR-1 and AGR-2 Fuel Fabrication

Having decided on its fuel form, the AGR program began two fuel fabrication campaigns: (1) one focused on laboratory-scale coating at ORNL to support the AGR-1 testing program; and subsequently, (2) a second at engineering scale at BWXT to support the AGR-2 testing program. The decision was made to initiate activities at laboratory scale for two primary reasons: (1) the 15-year hiatus in producing TRISO fuel in the U.S. resulted in the need to re-establish the capabilities, procedures, and expertise; and (2) to address the historical failure in the U.S. to produce fuel that would meet HTGR performance requirements as evidenced by poor irradiation performance in the commercial MHTGR and NP-MHTGR programs in the early 1990s. Fuel fabrication development activities for the AGR program have spanned 15 years. Laboratory-scale equipment was used for process development to reduce the time and cost to complete the tests from feedstock consumables to waste generation. As the program progressed, aspects of fuel fabrication operations graduated from laboratory-scale to engineering-scale equipment.

This section describes the processes used to fabricate the AGR-1 and AGR-2 TRISO particles. ***The fabrication techniques are provided for information only and are not intended to limit fabrication methods used to achieve the actual TRISO fuel specification provided in Table 5-5.***

As discussed in the following sections, the kernels and coatings of the UCO particles manufactured and tested in AGR-1 and AGR-2 exhibited some degree of property variation and were fabricated under different conditions and at different scales with remarkably similar excellent irradiation and accident safety performance. Thus, there is some allowance in terms of the actual values for key critical characteristics of the kernels and coatings necessary to impart satisfactory performance, as long as the TRISO particles meet the specification of Section 5.3.6.

5.3.1 Kernel Production

Nuclear fuel kernels can be produced by either external or internal gelation where a uranium "broth" containing an acid-deficient uranyl nitrate (ADUN) solution reacts with high pH chemicals surrounding the droplet (external) or incorporated into the broth (internal) causing the ADUN to convert to a uranyl hydroxide gel. The AGR program focused its kernel fabrication efforts on internal gelation chemistry, using hexamethylenetetramine (HMTA) and urea as ammonium donors to affect the gelation when the broth droplets were warmed in an immiscible forming fluid. Carbon black was added to the broth, prior to gelation, as a carbon source to make uranium carbides. After forming, the gel spheres were aged in a collection pot to firm up the gels and then washed with ammonia water to ensure complete gelation and to remove residual reactants and soluble salts. The gel spheres were then air-dried and heat-treated at high temperature to form hard, dense ceramic UCO microspheres.

5.3.1.1 AGR-1 Kernels

The AGR-1 UCO kernels were fabricated by BWXT in accordance with the AGR-1 Fuel Product Specification [46]. The fuel kernels had a nominal enrichment of 19.7% ^{235}U and a nominal diameter of 350 μm.

5.3.1.2 AGR-2 Kernels

For AGR-2, the U.S. kernels were fabricated by BWXT in accordance with the AGR-2 Fuel Product Specification [47]. The UCO kernels had a nominal enrichment of 14% and a nominal diameter of 425 μm. Several changes were made to the fabrication processes to improve the chemistry, integrity, and density of the kernels relative to AGR-1. Differences in the fabrication are discussed in the following section.

The AGR-2 UO_2 kernels had a nominal enrichment of 9.6% and a nominal diameter of 500 μm to be comparable with historic German fuel particles and to contrast the performance of domestic UO_2 TRISO particles with that of UCO TRISO.

5.3.1.3 Diversity in Kernel Production

Target process parameters, given in Table 5-1, show the main changes made to the kernel fabrication processes. Although the same equipment was used for forming, washing, and drying the kernels, some additional changes were made that are not documented in the table. These include ancillary equipment enhancements, such as the broth mixer, which was replaced between AGR-1 and AGR-2 kernel fabrication campaigns along with the nozzle orifice sizes and pulsation parameters that were changed to produce the desired droplet sizes. The measured and calculated characteristics of the AGR-1 and AGR-2 kernels are given in Table 5-2. All quantified impurity levels in the kernels were less than the specified maxima and commonly below analytical detection limits; these data are not included in Table 5-2.

Table 5-1
Differences in kernel production parameters forAGR-1 and AGR-2 [48-50]

Process Parameters	AGR-1 UCO	AGR-2 UCO	AGR-2 UO_2
HMTA : U mole ratio	1.3	1.55	1.55
Urea : U mole ratio	1.3	1.3	1.55
Carbon : uranium atom ratio (broth)	1.0	1.1	–
Carbon source	Powder	Aq. suspension	–
Carbon dispersion phase	ADUN solution	HMTA solution	–
Forming temperature	56°C	56°C	60°C
Broth droplet diameter (nominal)	1140 μm	1365 μm	1630 μm
Broth feed rate	27 cc/min	40 cc/min	30 cc/min
Calcination gas	100% H_2	8% H_2 : 92% Ar	13% H_2 : 87% Ar
Sintering gas	60% CO : 40% Ar	60% CO : 40% Ar	13% H_2 : 87% Ar
Sintering temperature	1890°C	1890°C	1500°C
Sintering hold time	4 hours	1 hour	2 hours

Table 5-2
AGR-1 and AGR-2 kernel properties [48-52]

Kernel Properties	AGR-1 UCO	AGR-2 UCO	AGR-2 UO_2
Diameter	349.7 μm	426.7 μm	507.7 μm
Density (g/cm^3)	10.92	10.97	10.86
Fraction of theoretical density (%)	90.4	95.7	99.0
Aspect ratio (D_{max}/D_{min})	1.021	1.012	1.009
C:U atom ratio	0.325	0.392	–
O:U atom ratio	1.361	1.428	2.003
[O+C]:U atom ratio	1.685	1.818	2.003
Calculated UO_2 : UC : $UC_{1.86}$ (†)	68 : 31 : 01	71 : 16 : 12	100 : 00 : 00
Wt% U	90.06	89.46	87.97

†Assumes that only the listed compounds were present.

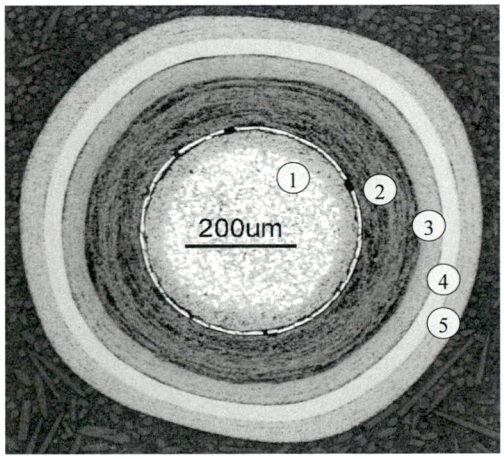

Figure 5-1
Sectioned TRISO fuel particle
Courtesy of Idaho National Laboratory and used with permission of Battelle Energy Alliance, LLC

5.3.2 TRISO Fuel Particles

An example of a UCO TRISO fuel particle from the AGR program is presented in cross section in Figure 5-1, with labels for the: (1) fuel kernel; (2) buffer layer; (3) IPyC layer; (4) SiC layer; and (5) OPyC layer.

During SiC deposition and the heat treatment of the compacted fuel form, the uranium monocarbide phase converts to the dicarbide within the kernel and releases elemental uranium that subsequently reacts at the buffer/kernel interface to form a skin of $UC_{1.86}$, evident as the thin, light ring between the kernel and the buffer layer in Figure 5-1.

5.3.2.1 TRISO Coating Deposition

The four coatings that comprise the TRISO fuel particle function the best in a reactor system if the coatings are sequentially deposited in the coater without an interruption that would necessitate keeping the bed fluidized for an extended time period or unloading the fuel. This is especially true for the IPyC and SiC layers, which provide the greatest fission product retention. Therefore, the coatings were sequentially applied without interruption. Uninterrupted coating is the baseline approach used by the successful German, Japanese, and Chinese programs and was adopted by the AGR program as well. The buffer is deposited by chemical vapor deposition from a mixture of acetylene and argon diluent. The inner and outer pyrolytic layers are deposited from a mixture of acetylene, propylene, and argon diluent. The SiC layer is deposited from methyltrichlorosilane (MTS) diluted with hydrogen and argon. Specifications are placed on the diameters, thicknesses, and densities of the kernel and coating layers; the sphericity of the kernel and coated particle; the stoichiometry of the kernel; the maximum anisotropy of the pyrocarbon layers; the microstructure of the SiC; and the acceptable defect levels for each layer. Statistical sampling techniques are used to demonstrate compliance with the specifications, usually at the 95% confidence level [53].

5.3.2.2 AGR-1 Particles

The AGR-1 UCO kernels were coated by ORNL, which also provided characterization data [54-57]. A baseline fuel and three variants were fabricated for AGR-1. These variants were purposely designed to explore a range of relevant process parameters to produce different physical values of key coating attributes, which had been the cause of the historical performance of U.S. TRISO fuel. Briefly, the baseline and variant fuels are described as:

- **Baseline**. Because of its excellent irradiation performance, coating process conditions used to fabricate historic German fuel were chosen as the starting point for the baseline fuel. Parametric studies refined these conditions for the specific coater used to coat AGR-1 fuel. This fuel was expected to perform successfully during irradiation.
- **Variant 1**. The IPyC coating temperature was increased relative to the baseline process for this variant. This change was expected to enhance the irradiation dimensional stability of the PyC, but with increased uranium dispersion. Also, the IPyC layer density was slightly lower than the baseline density.
- **Variant 2**. The IPyC coating gas fraction was increased relative to the baseline process for this variant. This change was also expected to enhance the irradiation dimensional stability of the PyC without significantly increasing uranium dispersion. Also, the IPyC layer density was slightly higher than the baseline density.
- **Variant 3**. The carrier gas composition for the SiC layer deposition was changed from hydrogen to an argon-hydrogen mixture, and deposition temperature was lowered. These changes were expected to change the microstructure of the SiC (including a finer grain size) and to reduce SiC defects.

The kernels were coated in a 2-in.-diameter retort tube. The 2-in.-diameter retort was selected, in part, because it facilitated fuel coating development studies without using a large amount of material resources or generating large quantities of waste.

5.3.2.3 AGR-2 Particles

The 425-µm UCO and 500-µm UO_2 kernels were coated and characterized by BWXT [49,50]. Based on the AGR-1 in-pile results available at the time, the AGR program decided the AGR-2 PyC coating would be applied using baseline conditions from AGR-1 and would use argon dilution during the SiC coating step, like AGR-1 Variant 3, for the best fluidization in the coater. The kernels were coated in a 6-in.-diameter retort increasing the coater capacity approximately 20-fold relative to AGR-1.

5.3.2.4 SiC Microstructure

The AGR-1 and AGR-2 fuel specifications did not include quantitative limits on SiC microstructure (e.g., SiC grain size). Instead, a visual standard was included that demonstrated a grain size considered to be excessively large. The visual standard comprises the two micrographs in Figure 5-2 below, with further guidance that the "specification will be met if the average SiC grain size of 3 coated particles is judged to be smaller than the average grain size shown in the visual standards." Thus the AGR program considered the example micrographs to represent an approximate upper bound on the acceptable grain size, with no specified lower bound. The AGR-1 test intentionally explored a range in grain sizes to evaluate the impact on performance. Quantitative data on AGR-1 and AGR-2 SiC grain size are provided in Table 5-3 and discussed in Gerczak et al (2016) [58].

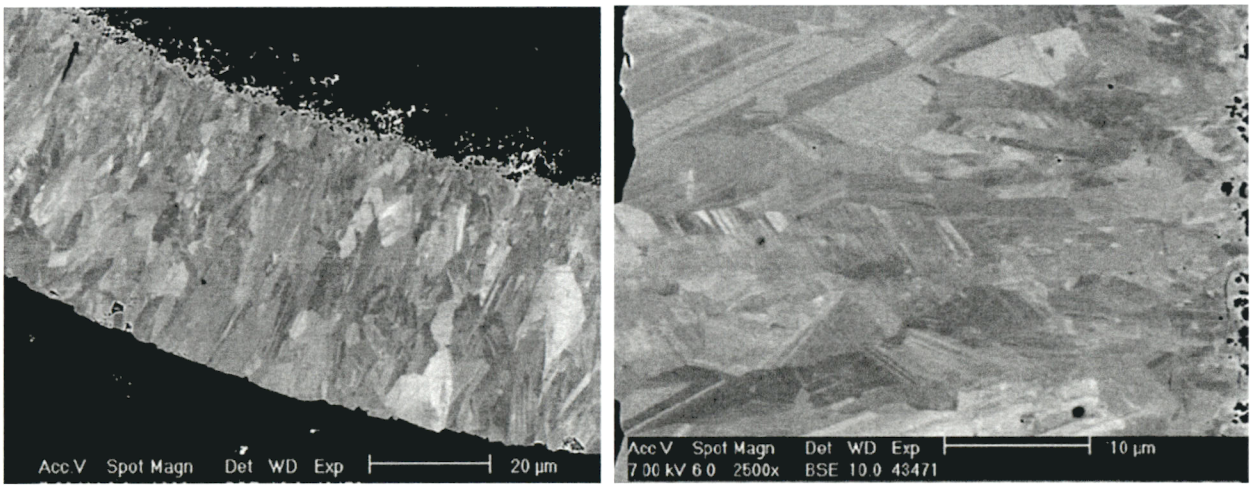

Figure 5-2
Visual standard for SiC microstructure used in the AGR-1 and AGR-2 fuel specifications
Courtesy of Idaho National Laboratory and used with permission of Battelle Energy Alliance, LLC

5.3.2.5 Diversity in TRISO Particle Properties

Properties of the resulting TRISO particles are given in Table 5-3.

Table 5-3
TRISO particle characterization data.

Layer Properties	AGR-1 [49,54-59]				AGR-2 [48,50,51,52,58-61]	
	Baseline	Variant 1	Variant 2	Variant 3	UCO	UO_2
Buffer Thickness (μm)	103.5	102.5	102.9	104.2	98.9	97.7
Density (g/cm^3)[a]	~1.10	~1.10	~1.10	~1.10	~1.04	0.99
IPyC Thickness(μm)	39.4	40.5	40.1	38.8	40.4	41.9
Density[a]	1.904	1.853	1.912	1.904	1.890	~1.89
BAF_o (True)[b]	1.015	1.009	1.015	1.020	1.024	1.023
SiC Thickness (μm)	35.3	35.7	35.0	35.9	35.2	37.5
Density (g/cm^3)	3.208	3.206	3.207	3.205	3.197	3.200
Aspect ratio[c]	–	–	–	–	1.037	1.034
Grain major axis (μm)[d]						
Twins	2.41	2.39	2.14	0.71	0.89	1.19
No Twins	5.82	5.10	5.29	1.29	1.67	2.37
OPyC Thickness (μm)	41.0	41.1	39.8	39.3	43.4	45.6
Density (g/cm^3)	1.907	1.898	1.901	1.911	1.907	1.884
BAF_o (True)[b]	1.013	1.009	1.012	1.014	1.018	1.015
Aspect ratio[c]	1.054	1.056	1.053	1.055	1.052	1.052
Missing OPyC	$\leq 9.7\times10^{-5}$	$\leq 9.7\times10^{-5}$	$\leq 9.6\times10^{-5}$	$\leq 9.7\times10^{-5}$	$\leq 1.90\times10^{-4}$	$\leq 5.8\times10^{-4}$

a. Layer density was not measured on batches with data preceded by a tilde (~). Values are inferred from similar runs.

b. The "true" Bacon anisotropy factor (BAF_o) is calculated from diattenuation (N) as follows: $BAF_o = (1+N)/(1-N)$. Original AGR-1 BAF_o anisotropies were calculated using a different conversion formula than later used for AGR-2. Data reported above use the same formula.

c. Aspect ratio is the ratio of major and minor radii. Measured only on the OPyC layer for AGR-1 fuel.

d. Grain major axis is reported only to indicate the impact on grain size of argon as a diluent gas combined with lower deposition temperatures. No correlation has yet been established between this property and SiC performance [58].

5.3.3 Sorting of Kernels and Particles

Various methods were employed to sort (separate by size and shape) the fuel kernels, TRISO-coated particles, and TRISO particles overcoated with resinated graphite powder. The methods described below have different sorting efficiencies and throughput rates. Table 5-4 is a summary of the methods employed to sort AGR-1 and AGR-2 materials at various stages of fuel fabrication.

Table 5-4
Sorting methods employed for AGR-1 and AGR-2 materials

Material	Sieved	Tabled	Sorted by Roller Micrometer
AGR-1 kernels	X	X	–
AGR-2 kernels	X	X	–
AGR-1 TRISO	–	X	X
AGR-2 TRISO	X	X	X
AGR-1 overcoated TRISO	X	X	–
AGR-2 overcoated TRISO	X	X	–

5.3.3.1 Sieving

Sieving is the most suitable method of sorting by size for full-scale production. Batch-wise sieving was employed for sorting of the AGR-1 and AGR-2 materials, but continuous methods could be employed for full-scale production. Sieving inherently sorts particles by the second largest dimension because the particle bed is in motion and particles can rotate to present different orientations to the apertures in the sieve, thus the longest axis is not always orthogonal to the plane of the sieve screen. The sieving rejected oversized and undersized kernels and particles and provided an opportunity to examine the reject fractions.

5.3.3.2 Tabling

Tabling is an operation where kernels or particles are passed over an inclined, vibrating plane to sort the materials by shape. The more spherical materials readily traverse the plane and are collected in product bins opposite the feed port. Non-spherical materials do not roll well and move more orthogonally to the flow of spherical material and are collected in reject bins. Tabling is most efficient and has its greatest utility when the bulk of the materials are highly spherical. Faceted particles, such as TRISO particles, are more difficult to sort by shape due to increased comingling of the product and reject streams.

5.3.3.3 Roller Micrometer Sorting

A roller micrometer consists of two cylindrical rods sloping away from the feed point and slightly diverging. The two rods rotate in opposite directions and away from the center line so as to roll the particles as they roll toward the widest and lowest end. Particles are sorted by their minimum dimension. Because particles are sorted in a single-file, this process is time-consuming and thus less well-suited for large-scale production.

5.3.4 Fuel Compact Fabrication

Whether a cylindrical compact, a pebble, or another fuel form is to be pressed, a host matrix is needed to provide the structural integrity and thermal conductivity of the fuel form while benefiting the reactor physics. Graphite is a suitable medium, as it can provide exceptional high-temperature strength and good dimensional stability, and it moderates neutrons. A binder is needed to get graphite powders to remain in the compacted shape and to achieve the needed structural strength and integrity of the fuel form. Phenolic resins bind the graphite particles well, pyrolyze to an amorphous carbon phase, and are readily available in a highly pure form.

After coating, AGR-1 and AGR-2 particles were formed into right cylindrical compacts at ORNL. Prior to compacting, the AGR fuel particles were overcoated with resinated graphite powder. The resinated graphite powder becomes the compact matrix upon compaction and heat treatment. This overcoat also served to prevent particle-to-particle contact and to help achieve the desired volumetric packing fraction of fuel particles within the compacts. Resinated graphite powder was added to the die bodies before charging the overcoated TRISO particles and again afterward to form an unfueled end cap on the compacts as a precaution against damaging TRISO particles during compaction. The compacts were nominally 25 mm in length and 12.3 mm in diameter with fuel-free end caps of matrix material approximately 1.5 mm thick for AGR-1 and 0.5-mm thick for AGR-2. These end caps ensured smooth, protected surfaces that helped to prevent fuel particle damage during handling. The end cap thickness was reduced for AGR-2 and eliminated for subsequent AGR irradiation experiments. The AGR-1 compacts were pressed at room temperature using a single-acting die and a Carver press. The AGR-2 compacts were pressed at approximately 70°C using a Promess press, but utilized a die with a floating die body to function more like a double-acting press. The overcoated TRISO particles were pre-treated in a methanol atmosphere to soften the resin and make the overcoat more malleable.

5.3.5 Quality Controls and Statistical Methods for Characterizing Fuel

Quality controls and statistical methods for characterizing unirradiated HTGR fuels from the fuel kernels to the final fuel form are outside of the scope of this report. Considerable information on analytical methods for characterizing fuel is available in *Characterization and Advanced Quality Control Techniques* in IAEA-TECDOC-1674, *Advances in High Temperature Gas Cooled Reactor Fuel Technology* [62].

Guidance on general statistical methods is available in report INL/EXT-05-00349, *Statistical Methods Handbook for Advanced Gas Reactor Fuel Materials* [63]. Experiment-specific guidance is found in the *Statistical Sampling Plan for AGR Fuel Materials* [53] and *Statistical Sampling Plan for AGR-2 Fuel Materials* [64].

5.3.6 Key Property Ranges Observed in AGR-1 and AGR-2 TRISO-Coated Particles

Table 5-5 summarizes the key coating layer properties for the particles used in the AGR-1 and AGR-2 irradiation experiments. Note that selection of key particle properties for this list is influenced, in part, by extensive thermomechanical modeling of particle performance and sensitivity studies to determine which properties have the greatest impact on particle failure probability, as well as historic TRISO fuel experience. The data represent a combination of the

values from all of the AGR-1 and AGR-2 fuel types (fuel types include AGR-1 Baseline, its three variants, and AGR-2 UCO particles), except in the case of aspect ratio.

Two ranges of values are given for each property, one drawn from confidence intervals on the means and one drawn from tolerance intervals for the populations. Ranges for the mean were drawn from the collection of two-sided 95% confidence intervals on the mean for each fuel type. Ranges characterizing the tails of the property distributions were drawn, except as noted, from the collection of 95% confidence – 98% coverage tolerance intervals for each fuel type. The approach is illustrated schematically for three particle populations in Figure 5-3. Note that while each of the type-specific intervals represent assumed normal distribution, the ranges provided in Table 5-5 are not equivalent intervals for the entire population of particles. Nonetheless, the ranges given are considered useful bounds owing to the considerable overlap in the individual distributions and the fact that each individual population was well represented in the irradiation experiments.[16]

When comparing a fuel population to the AGR-1 and AGR-2 fuels, the tolerance limit extrema in Table 5-5 are considered of greater importance than the mean confidence limits, because the tolerance limits serve to define the fraction of particles with the most extreme property values. The mean value for a fuel population may be outside of the range in Table 5-5 while the 95%/98% tolerance limits still reside inside of the corresponding limits in Table 5-5. This would indicate a similar or smaller fraction of particles with properties outside of these bounds, and therefore it would be straightforward to conclude that the fuel performance would be equivalent to the AGR fuel in similar irradiation conditions.

The values in Table 5-5 are not intended to define a comprehensive envelope of TRISO fuel that is expected to have acceptable performance. The data characterize the range of properties for particles that performed well during the AGR-1 and AGR-2 irradiations, but do not define the only ranges or combination of ranges that would perform well under these irradiation conditions or under service conditions proposed by fuel fabricators and reactor designers. The values are provided to facilitate comparison of other TRISO fuel populations to the fuel tested in the AGR-1 and AGR-2 irradiations. Comparative analysis of another population possessing particle properties that deviate from those in Table 5-5 will vary in complexity based on the specific properties in question (not all particle properties impact fuel performance in the same manner), the magnitude of the deviations (including which end of the distribution exceeds the limits in Table 5-5), and the intended irradiation conditions. More detailed data describing AGR-1 and AGR-2 particles are available for comparisons between AGR fuel and other TRISO fuel populations (see the references listed in Table 5-3). Ultimately it will be up to an applicant to provide a justification for applying AGR-1 and AGR-2 particle performance results to a TRISO fuel population that deviates from the AGR-1 and AGR-2 fuel properties.

[16] Additional details on the calculation of the ranges in Table 5-5 are given in A. Mack, Characterization of AGR-1 and AGR-2 UCO TRISO Particle Layer Property Distributions, ECAR-5254, Idaho National Laboratory, November 2020.

Table 5-5
Coating layer property ranges for irradiated AGR-1 and AGR-2 UCO particles

Particle Property	95% Confidence Interval Extrema	95%/98% Tolerance Limit Extrema
Buffer thickness (μm)	96.5 – 105.2	75.2 – 124.7
IPyC thickness (μm)	38.6 – 41.1	32.4 – 47.6
SiC thickness (μm)	34.8 – 36.2	30.6 – 41.2
OPyC thickness (μm)	39.1 – 44.3	33.6 – 51.6
Buffer density (g/cm^3)	1.04 – 1.11 [a, b]	NA
IPyC density (g/cm^3)	1.84 - 1.92 [b]	1.808 – 1.958 [b]
SiC density (g/cm^3)	3.196 – 3.209	3.191 – 3.217
OPyC density (g/cm^3)	1.878 – 1.924	1.850 – 1.949
IPyC anisotropy (BAF_{True}) [c]	1.024 [d]	1.036 [d]
OPyC anisotropy (BAF_{True}) [c]	1.018 [d]	1.030 [d]
Aspect Ratio	1.057 [d, e] 1.040 [d, f]	1.102 [d, e] 1.068 [d, f]

a. Range of measured means only. No confidence intervals available.
b. Indirectly measured by analysis of interrupted batches (AGR-1) or comparable batches (AGR-2 buffer density).
c. $BAF_{True} = (1+N)/(1-N)$, where N is the optical diattenuation.
d. Upper bound of 95% confidence interval or 95% confidence - 99% coverage tolerance interval, as appropriate.
e. AGR-1, OPyC layer
f. AGR-2, SiC layer

It should be noted that the ranges given in Table 5-5 are narrower than the ranges specified in the AGR-1 and AGR-2 fuel specifications (means and 1% critical limits) (see Appendix C for AGR fuel specification information). TRISO particle performance is primarily defined by the probability of in-service coating layer failure. This behavior varies with key fuel properties, and the impact on coating layer fracture differs in magnitude for each property. Within a certain range of values, the impact on fuel performance will be negligible. The AGR fuel specification ranges were determined based on past performance demonstrations and on thermomechanical modeling of fuel performance to determine ideal property values as well as to identify the extremes of property distributions where fuel performance would be expected to begin to degrade appreciably. The AGR fuel specification 1% critical limits are established to be within this range; appreciable increases in fuel failure would only be expected outside of these bounds. Beyond these bounds, the particle failure probability increases, with performance generally becoming worse as the property value gets further from the mean. Thus, a particle in a fuel population could reside outside of the Table 5-5 ranges, but still be within the AGR fuel

specification and be expected to perform well under the AGR irradiation conditions, based on the amassed knowledge of TRISO fuel performance over the last several decades.

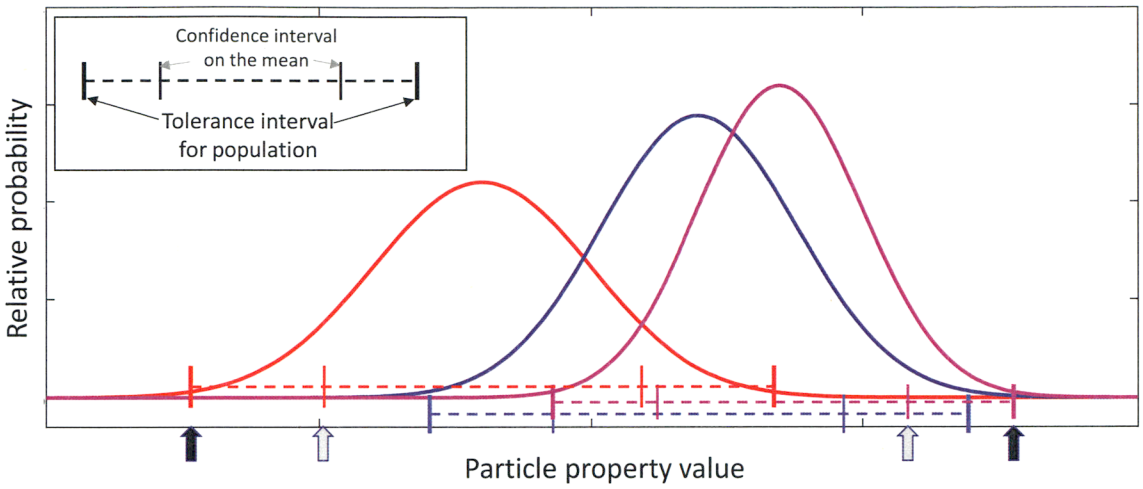

Figure 5-3
Schematic diagram illustrating the selection of confidence intervals extrema and tolerance intervals extrema from a hypothetical set of populations each characterized by a normal distribution, with confidence intervals and tolerance intervals denoted. Area under each curve represents the relative number of particles per batch/fuel type. Gray arrows identify the extrema of the confidence intervals on the mean for each population. Black arrows identify the extrema of the tolerance intervals.

As noted in Section 4.2, because the kernel is thermomechanically decoupled from the coating layers, there is not a *unique* set of kernel specifications that are critical to successful TRISO fuel as long as the scaling discussed in Section 4.2 is considered. Historically, a broad range of fissile and fertile kernels in a variety of chemical forms have been irradiated successfully around the world.

Thermochemical calculations have been performed previously on the relative stability of the uranium oxide and carbide phases in UCO fuel, as well as the stability of fission product oxides and carbides. Homan et al. [25] presented the graphic shown in Figure 5-4, which indicates the range of burnup over which the various phases are stable given a specific starting UO_2/UC_2 content in the kernels at 1800 K. The results indicate that at UC_2 content as low as 10% the UC_2 phase will still persist and limit the formation of CO gas up to ~18% FIMA. Beyond this burnup, the oxide/carbide equilibrium for strontium establishes the oxygen potential, and thereafter the equilibrium in the zirconium system. On the other hand, at UC_2 content as high as 80%, the rare earth fission products are still retained in the kernel as oxides. This demonstrates the wide range of UO_2/UC_2 ratios that maintain effectiveness at (a) limiting CO gas formation and (b) promoting the formation of rare earth oxides over the formation of rare earth carbides in order to increase retention of rare earths in the kernel.

Subsequent thermochemical studies have suggested that UC_x content as low as 5.5% (C:U ≈0.1) is sufficient for acceptable performance of low enriched uranium UCO TRISO fuel up to 16% FIMA [65]. The AGR program chose to target about 30% uranium carbide in their kernel fabrication to provide ample carbide phase to meet a burnup of ~20% FIMA while experiencing negligible CO gas formation.

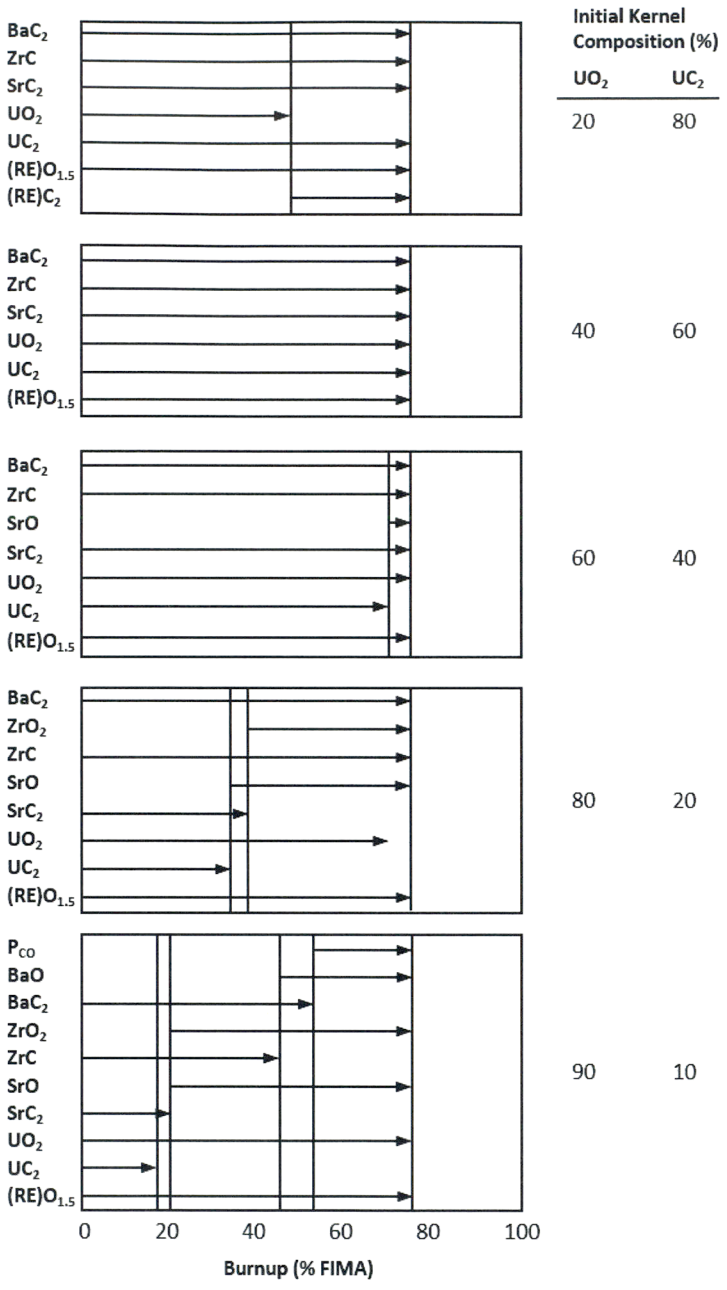

Figure 5-4
Phases present in UCO kernels as a function of starting UC_2 content and burnup
Reproduced from Homan et al. (1977) [25] with permission of the American Nuclear Society, http//www.ans.org/.

Section 4.2.6 introduced the concept of a SiC stress metric (σ), defined in Eq. 4-1. The stress in the SiC layer is proportional to the volume of kernel (V_k) and buffer (V_b), the inner radius (r_{SiC}) and thickness (t_{SiC}) of the SiC layer, and the intended peak burnup (B). Values for this tensile stress metric for various TRISO particle designs are provided in Table 4-2 based on nominal particle dimensions, showing that historic particle design has sought to maintain a value similar to that of the German reference UO_2 TRISO particle.

In the stress metric, the four fundamental geometric terms are the kernel radius, buffer thickness, IPyC thickness, and SiC thickness. Using the as-fabricated fuel characterization data, bounding values for the stress metric, σ, in Eq. 4-1 were calculated to demonstrate the particle-to-particle variability in the AGR-1 and AGR-2 fuels. The distribution of σ for particles with the greatest burnup achieved in each particle type was calculated using Monte Carlo simulation, based on sample means and standard deviations of kernel diameter and layer thicknesses of each particle type. In simulation, the distribution of the quantity of interest is estimated by generating normal random deviates for each of the uncertain terms involved in its calculation and calculating the resultant stress metric for each set of those values. The 1st, 50th, and 99th percentiles of the simulated stress metric distribution, for compacts with highest burnup for each fuel type, are provided in Table 5-6.

The data in Table 5-6 demonstrate that (a) there is a significant range in values for the stress metric based on statistical variations in particle geometry, and (b) the peak values (99^{th} percentile) based on this analysis are within the range of values (0.643-0.816) listed in Table 4-2 representing values for previous and current fuel designs.

Table 5-6
Mean and percentiles of the stress metric calculated from the Monte Carlo simulation. Minimum, median and max are the 1%, 50%, 99% percentile values of the generated stress metric values.

Test	SiC Stress Metric σ			
	Mean	Distribution quantiles		
		1%	50%	99%
AGR-1	0.570	0.440	0.566	0.742
AGR-2	0.623	0.485	0.618	0.810

6
AGR-1 AND AGR-2 IRRADIATIONS

The irradiation performance of the TRISO fuel produced in the 1990s compared to the successful German program led to a broad review of all aspects of fabrication and testing of TRISO fuel, providing lessons learned for future TRISO fuel work in the United States [66]. The review suggested changes in the fabrication process to improve coating performance, recommended a reduction in the level of acceleration in fuel irradiations, and urged an expansion of PIE to fully characterize the fuel following irradiation and/or accident safety testing. With this historical backdrop, the following objectives and goals were defined for AGR-1 and AGR-2.

- **AGR-1**. The goal was to fabricate different types of UCO TRISO fuel particles using a 2-in. laboratory-scale coater at ORNL under a set of systematic, well-characterized coating conditions. As discussed in Section 5.3.2.2, a baseline fuel particle composite and three variant fuel particle composites were fabricated. The variants included two particle composites coated using different IPyC coating conditions and one particle composite coated using different SiC coating conditions. In the area of irradiation, a key objective was to gain experience with multi-capsule test train design, fabrication, and operation to reduce chances of operational problems in subsequent test trains. Such types of capsules had been used successfully in Europe to support German TRISO fuel qualification. Another goal was to obtain early data on irradiated fuel performance and support development of a fundamental understanding of the relationship between the fuel fabrication process, fuel product properties and irradiation performance. If the fuel performance under irradiation was acceptable, there would be ample irradiated UCO fuel for accident simulation testing (that is, heating tests) and other PIE activities. In terms of accident testing, two separate furnaces were established at INL and ORNL to conduct long-term high-temperature heating tests to simulate accident performance similar to the German program. In addition, significant infrastructure and capabilities were established at hot cells at both laboratories to: (1) characterize particles after irradiation and accident heating; (2) establish fission product mass balances; and (3) search for and recover any failed or degraded particles to understand the causes for such behavior.
- **AGR-2**. The objective was to demonstrate the performance of TRISO-coated UCO particles fabricated in a 6-in. engineering-scale coater. The irradiation capsule design for AGR-2 was essentially the same as demonstrated in AGR-1; it had six independently monitored and controlled capsules in a test train. Three capsules contained UCO fuel. Two of these were irradiated under normal conditions, while one UCO capsule was operated with a maximum time-average temperature of about 1360°C as a performance margin test of the fuel. The remaining three capsules tested UO_2 TRISO fuel (one containing U.S.-manufactured UO_2 particles, while the other two contain particles from France and South Africa). Although the focus of this report is on the performance of TRISO-coated UCO particles, the results on the UO_2 performance in AGR-2 are also provided as a benchmark given it is the historic fuel form used around the world (that is, Germany, China, and Japan).

6.1 Capsule Design and Operation

AGR-1 and AGR-2 were irradiated in the 38.1 mm (1.5 in.) diameter east and west large B positions (B-10 and B-12), respectively, at the INL ATR [59,61]. A cross-sectional view of the ATR core indicating the location of the east large B position is displayed in Figure 6-1. *A priori* physics calculations [67] showed anticipated very high-temperature reactor end-of-irradiation conditions (that is, burnup to about 20% FIMA and maximum fast neutron fluence of 5×10^{25} n/m^2, E >0.18 MeV) were best matched by the conditions obtained from irradiation in these large B positions after about 550 to 600 days of irradiation.

Conducting irradiations in these locations results in a slight acceleration in the accumulation of burnup and fast fluence compared to that expected in historic HTRs. Target burnups can be reached in about 550 to 600 effective full power days in the ATR compared with 1000 days in historic HTRs. The actual acceleration factors observed in AGR-1 and AGR-2 are less than 2, consistent with the bulk of the historic German irradiation experience and much slower than the bulk of the historic U.S. irradiations that were highly accelerated (x5-10) [66]. The effect of accelerated irradiation has been examined based on the current understanding of TRISO fuel performance. Accelerated irradiations can lead to higher peak temperatures in the fissile kernels of coated particles and for very high acceleration factors, the temperatures can be 100 to 500°C higher depending on the design of the coated particle [68]. However, the more modest accelerations of the AGR irradiation and the historical German testing show little to no effect on fuel performance. Furthermore, as discussed later in this report, the PIE of AGR-1 and AGR-2 TRISO fuel show no indication of any potential incipient failure that could have occurred had the time at temperature been longer as would be the case in a real time irradiation.

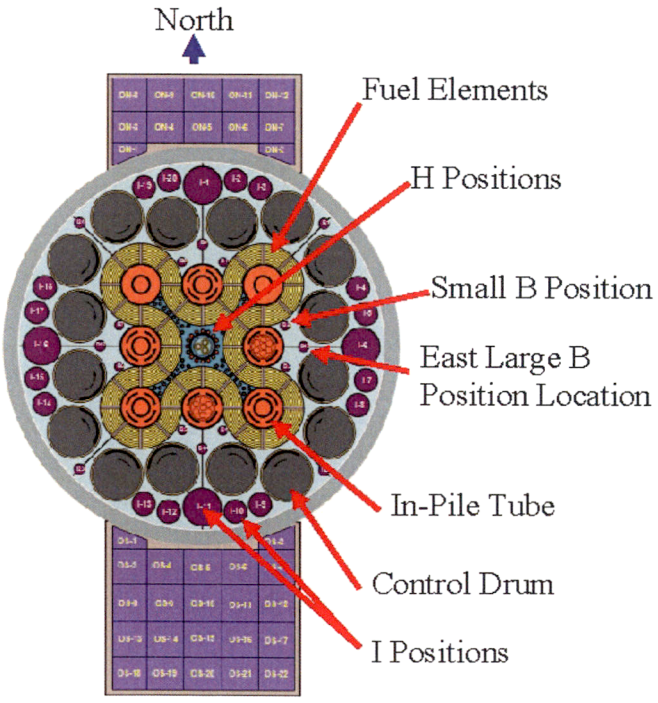

Figure 6-1
ATR core cross section
Courtesy of Idaho National Laboratory and used with permission of Battelle Energy Alliance, LLC

The AGR-1 and AGR-2 test trains were multi-capsule, instrumented lead experiments with very similar design. Each test train contained six capsules, each independently controlled for temperature and independently monitored for fission product gas release. An axial view of the test train is illustrated in Figure 6-2. Each capsule was 152.4 mm (6 in.) long and contained 12 fuel compacts arranged in three vertical stacks, with each stack containing four compacts. Figure 6-3 shows a cutaway view of an AGR-1 capsule illustrating the arrangement of the three compact stacks and showing the hafnium shroud used to suppress flux on the west side of the capsule.

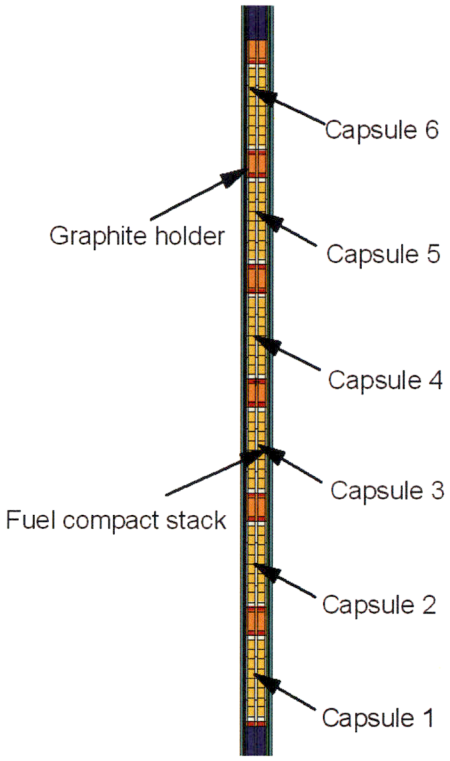

Figure 6-2
Axial schematic of the AGR-1 capsules
Courtesy of Idaho National Laboratory and used with permission of Battelle Energy Alliance, LLC

Independent gas lines routed a mixture of helium and neon gases through each of the six capsules to provide temperature control and to sweep released fission product gases to the fission product monitoring system (FPMS). Temperature control was based upon temperature feedback from the thermocouples (TCs) in each capsule and was performed by varying the sweep gas composition (between 100% helium for high conductivity and 100% neon for low conductivity). This blending of sweep gases before the gas enters the test train could be accomplished either automatically (by a computerized mass flow controller) or manually. The arrangement of the gas lines can be seen in the three dimensional (3-D) rendering of a test capsule shown in Figure 6-3.

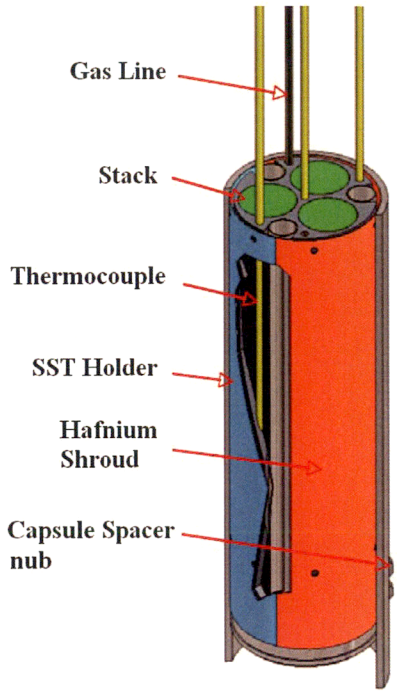

Figure 6-3
Three-dimensional cutaway rendering of single AGR-1 capsule
Courtesy of Idaho National Laboratory and used with permission of Battelle Energy Alliance, LLC

A horizontal capsule cross section at the top of the AGR-1 test train is shown in Figure 6-4. AGR-2 was similar in design but was a mirror image of AGR-1 since it was irradiated in an identical position on the other side of the ATR core in the west large B position.[17] In both experiments, the compacts were placed inside a boronated graphite sleeve. The boron allowed a reduction in heat generation early in the experiment to provide more uniform heating (compared to the exponential drop in heating expected in the case of no boron as the fuel was completely depleted of ^{235}U) and better thermal control of the experiment.

Each capsule contained only one fuel type or variant. In AGR-1, baseline fuel was irradiated in Capsules 6 and 3, Variant 1 in Capsule 5, Variant 2 in Capsule 2, and Variant 3 in Capsules 1 and 4. In AGR-2, U.S. UCO fuel was irradiated in Capsules 2, 5, and 6; U.S. UO_2 fuel in Capsule 3; French UO_2 fuel in Capsule 1; and South African UO_2 fuel in Capsule 4. The capsules are numbered consecutively from the bottom (Capsule 1) to the top (Capsule 6). Fuel compacts are identified by their location in the test train using a three-digit (*X*-*Y*-*Z*) nomenclature, where *X* refers to the capsule number, *Y* refers to the axial level within the capsule (Level 4 is at the top of the capsule and Level 1 is at the bottom), and *Z* refers to the stack number.

[17] Note that the AGR-2 test train was removed from the core several times to avoid higher-power cycles, and was irradiated for one cycle in the I-24 position. Details can be found in Reference 60.

The FPMS continuously measured the sweep gas from each capsule to provide an indicator of fuel irradiation performance [69]. Spectrometer detector systems measured the concentrations of various krypton and xenon isotopes in the sweep gas from each capsule. Eight-hour counting intervals were used to measure the concentrations of ^{85m}Kr, ^{87}Kr, ^{88}Kr, ^{89}Kr, ^{90}Kr, ^{131m}Xe, ^{133}Xe, ^{135}Xe, ^{135m}Xe, ^{137}Xe, ^{138}Xe, and ^{139}Xe.

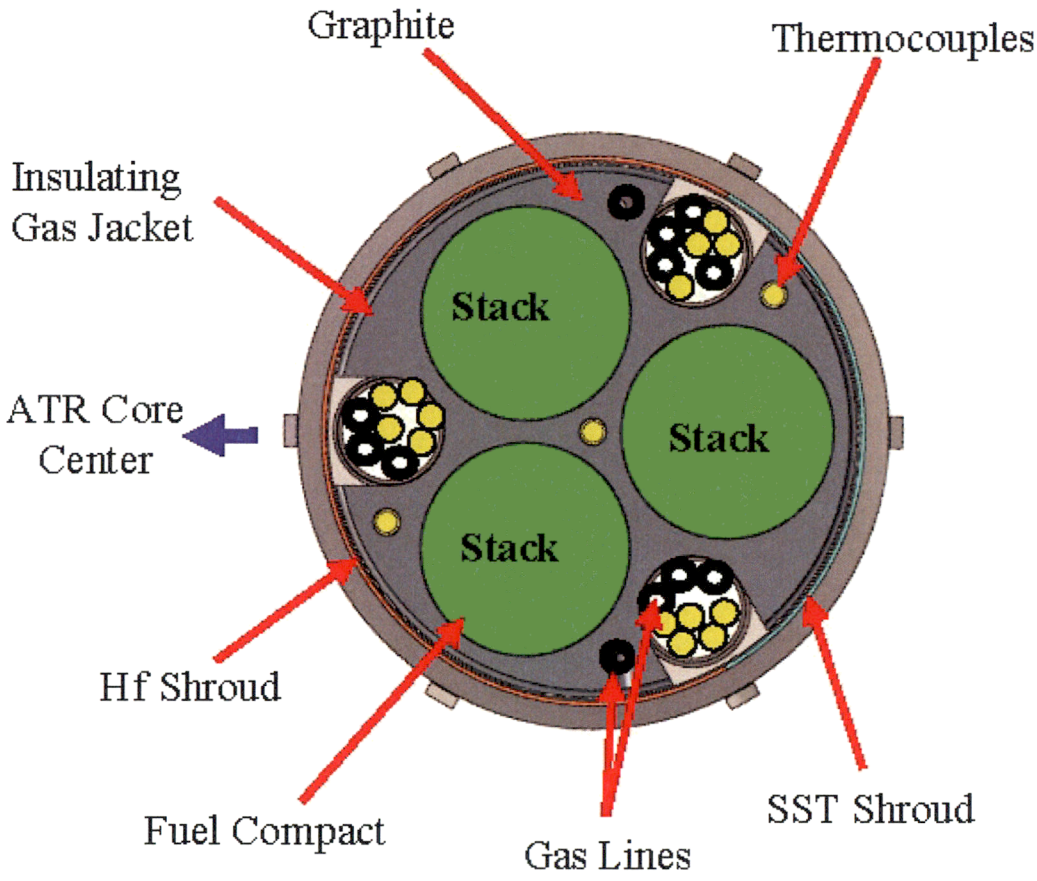

Figure 6-4
Horizontal cross section of an AGR-1 experiment capsule
Courtesy of Idaho National Laboratory and used with permission of Battelle Energy Alliance, LLC

The FPMS incorporated seven individual monitoring systems: one for each of the individual capsule effluent lines, and one that could monitor any individual effluent line or any combination of the six lines. This seventh monitor was primarily provided as a backup unit capable of providing effluent line monitoring should any of the primary monitoring systems fail. Each monitor consisted of a high purity germanium (HPGe) detector-based, gamma-ray spectrometer, and a thallium-activated, sodium iodide (NaI (Tl)) scintillation detector-based total radiation detector (often termed the "gross" radiation detector). The gross detectors were able to detect the failure of individual TRISO particles, while the gamma-ray spectroscopy was used for isotopic quantification of the noble gas release. These detector units are located in the ATR-2C secondary cubicle. Figure 6-5 illustrates the flow path used for both the AGR-1 and AGR-2 irradiations.

The sweep gas from each test capsule was routed via sampling lines to the monitoring station associated with that capsule. The sample lines, valves, and filters are predominately contained in the 2C primary cubicle. The sample lines have only two short, shielded segments in the 2C secondary cubicle. These short segments run through the gross detector monitoring station and into the HPGe spectrometer shield.

Each gross detector monitoring station (seven stations implemented) incorporates a Ø25 × 25 mm NaI (Tl) scintillation detector viewing a 25-mm-long segment of the capsule effluent line just before its entry into the HPGe spectrometer shield. The scintillation detector counting rate is monitored using a computer-controlled multi-channel scaler.

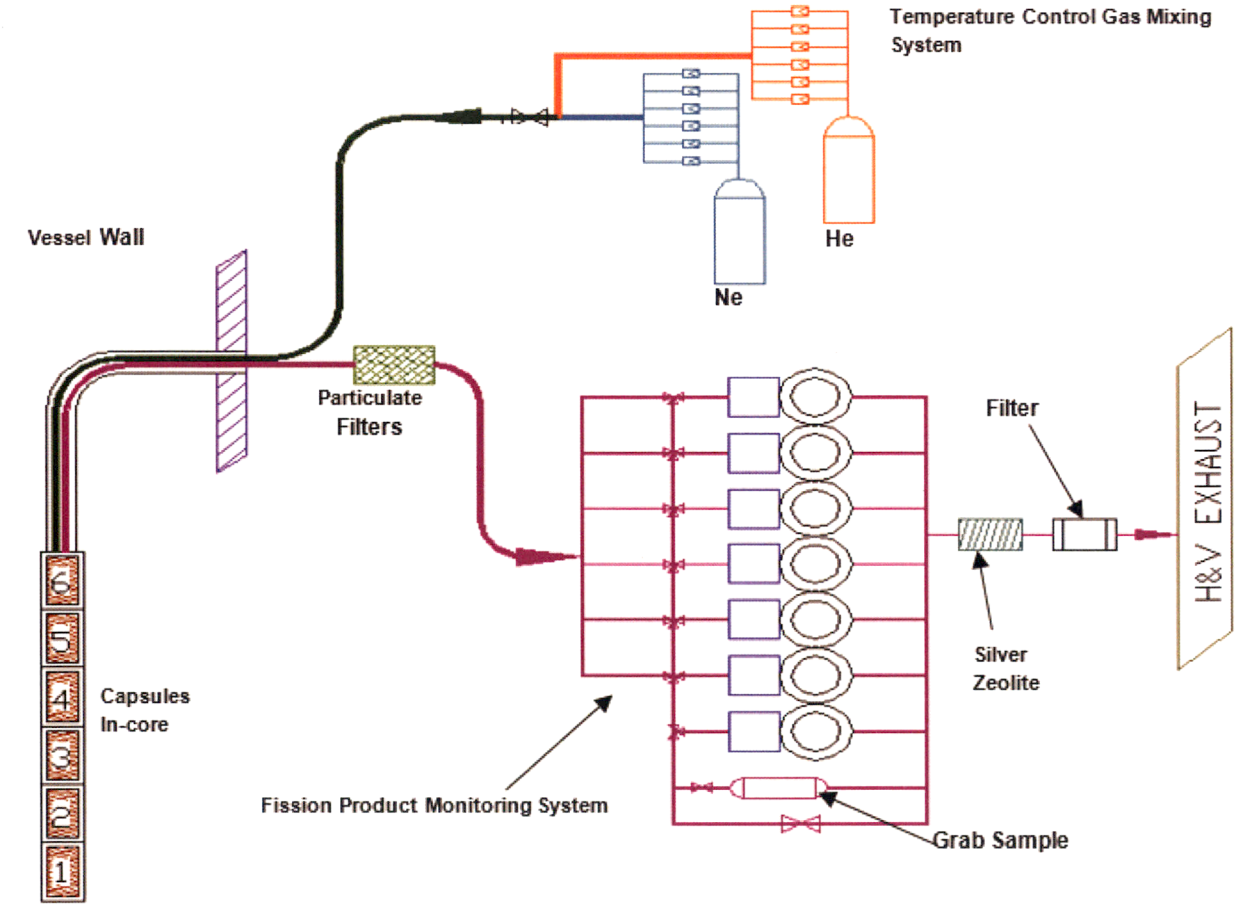

Figure 6-5
AGR-1 and AGR-2 experiment gas flow path
Courtesy of Idaho National Laboratory and used with permission of Battelle Energy Alliance, LLC

6.2 Fission Rate, Burnup, and Fast Fluence

Neutronics analysis of the experiments was performed using JMOCUP, a depletion calculation code developed at INL combining the continuous energy Monte Carlo N-Particle (MCNP) transport code [70] and the depletion code ORIGEN 2.2 [71]. The JMOCUP depletion methodology was used to model and deplete the AGR-1 and AGR-2 TRISO fuel compacts.

Figure 6-6 and Figure 6-7 show the calculated capsule-average heat generation rate in the AGR-1 and AGR-2 compacts versus time in effective full power days (EFPDs). The compact fission power densities varied between ~50 and 150 W/cc for both irradiations, but in rare cases exceeded 150 W/cc at the end of some of the irradiation cycles. The general trend shared by each capsule is an increase over the first several cycles as the boron in the graphite was depleted, followed by a leveling-off over the remaining cycles.