Enclosure 2

University of Illinois Urbana-Champaign High Temperature Gas-cooled Research Reactor: Fuel Qualification Methodology

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University of Illinois Urbana-Champaign High-Temperature Gas-cooled Research Reactor: Fuel Qualification Methodology

TOPICAL REPORT

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EXECUTIVE SUMMARY

This Topical Report (TR) describes the fuel qualification methodology for the Ultra Safe Nuclear Corporation (USNC) Micro-Modular Reactor (MMR™). The fuel for the MMR consists of micro-encapsulated tristructural isotropic (TRISO) fuel particles embedded in a ceramic matrix to form a Fully Ceramic Micro-encapsulated (FCM®) fuel pellet.

The fuel qualification methodology of the MMR FCM fuel is based on international and U.S. operating experience with TRISO fuel particles, including the extensive irradiation testing and post-irradiation safety testing of TRISO fuel particles by the U.S. Department of Energy (DOE) Advanced Gas Reactor (AGR) Fuel Development and Qualification Program.

Safety and fission product retention for normal operations and accident conditions is achieved through a combination of a well-studied fuel form (TRISO) and a unique, additional barrier (FCM) to provide defense-in-depth. TRISO particle fuel performance benefits from substantial and growing international and U.S. experience being used within the parameters previously reviewed and approved by the U.S. Nuclear Regulatory Commission (NRC). In addition to the fission product barriers present in TRISO fuel, the MMR FCM fuel incorporates an additional high-integrity silicon carbide (SiC) barrier. The additional SiC barrier is fabricated using a process developed to minimize effects detrimental to TRISO fuel particle performance. The FCM fuel form, including the embedded TRISO particles and outer SiC barrier, is designed to provide excellent fission product retention over 20 years of operation. Because FCM is a new fuel form with limited operational experience regarding its irradiation performance, FCM fuel pellets will be tested through the fuel qualification program described in this TR. Fuel fabrication specifications will identify characteristics and processes so that the fuel manufactured for use in the MMR is bounded by the FCM fuel qualification envelope.

Activities included in USNC's fuel qualification program include:

- Development of fuel product specifications for TRISO particles and FCM pellets
- Demonstration of fuel manufacturing and quality control processes capable of consistently meeting specifications
- Testing and characterization of unirradiated fuel and materials
- Fuel pellet irradiation tests in material test reactors
- Post-irradiation safety testing of irradiated fuel pellets to measure performance in simulated accident conditions
- Post-irradiation examination of fuel pellets after irradiation testing and after post-irradiation heat-up tests to determine fuel performance and material properties of irradiated fuel pellets
- Development and validation of fuel performance models

A fuel surveillance program will be implemented per ANSI/ANS-15.1, Sections 3.3(5) and 3.7.1(2), i.e., standards developed by the American National Standards Institute (ANSI) and the American Nuclear Society (ANS). The fuel surveillance program will perform online monitoring to ensure that fission product release remains within operational limits.

UNSC's fuel qualification program provides reasonable assurance that the MMR FCM fuel design can operate with a low failure rate and a level of fission product release consistent with the design basis analysis. The results of the fuel qualification program will be submitted through a licensing application to the U.S. NRC. It is requested that the U.S. NRC review Sections 5.1, 5.2, 5.3, and 5.4 and the acceptance criteria listed in Section 5.6 of this report and approve the fuel qualification methodology described therein. Following NRC approval of this fuel qualification methodology TR, it is expected that meeting the acceptance criteria in Section 5.6 of the TR by a license applicant will qualify the fuel for use as part of its application.

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ABBREVIATIONS & ACRONYMS

This list contains the abbreviations and acronyms used in this document.

Abbreviation or Acronym	Definition
AC	Alternating Current
AGR	Advanced Gas Reactor
AM	Additive Manufacturing
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
ASQ	American Society for Quality
ATR	Advanced Test Reactor
AVR	Arbeitsgemeinschaft Versuchsreaktor
BAF	Bacon Anisotropy Factor
BDBA	Beyond Design Basis Accident
BISO	Bistructural Isotropic
BP	Burnable Poison
CVD	Chemical Vapor Deposition
CVI	Chemical Vapor Infiltration
DBA	Design Basis Accident
DBE	Design Basis Event
DOE	[U.S.] Department of Energy
DPA	Displacements Per Atom
EAB	Exclusion Area Boundary
ECCS	Emergency Core Cooling System
EFPD	Effective Full Power Day
EPRI	Electric Power Research Institute
EU	Enriched Uranium
FIMA	Fissions per Initial heavy Metal Atom
FCM®	Fully Ceramic Micro-encapsulated
FQAF	Fuel Qualification Assessment Framework
FSAR	Final Safety Analysis Report
FSV	Fort St. Vrain
GIF	Generation-IV (Gen-IV) International Forum
HALEU	High-Assay Low-Enriched Uranium (i.e., enriched 5% to 20% in ^{235}U, exclusive)
HFR	High Flux Reactor
HMTA	Hexamethylenetetramine
HTGR	High-Temperature Gas-cooled Reactor
IAEA	International Atomic Energy Agency
IPyC	Inner PyC
LBE	Licensing Basis Event
LEU	Low-Enriched Uranium (i.e., enriched 0.72% to 4.95% in ^{235}U)
LWR	Light Water Reactor

MITR	Massachusetts Institute of Technology Reactor
MHR	Modular Helium Reactor
MHTGR	Modular High-Temperature Gas-cooled Reactor
MHTGR-DC	Modular High-Temperature Gas-cooled Reactor - Design Criteria
MMR™	Micro-Modular Reactor
MTR	Material Test Reactor
MTS	Methyltrichlorosilane
MW(e)	MW electric
MW(t)	MW thermal
NGNP	Next Generation Nuclear Plant
NRC	[U.S.] Nuclear Regulatory Commission
{{	}}
OPyC	Outer PyC
O/U, O/C	Oxygen-to-uranium atomic ratio, oxygen-to-carbon atomic ratio
PDC	Principal Design Criteria
PSAR	Preliminary Safety Analysis Report
PIE	Post-Irradiation Examination
PSER	Preapplication Safety Evaluation Report
PyC	Pyrolytic Carbon (aka pyrocarbon)
R/B	Release rate to Birth rate ratio
RCCS	Reactor Cavity Cooling System
RCSS	Reactivity Control and Shutdown System
RG	Regulatory Guide
RN	Radionuclide
SA	Sensitivity Analysis
SAFDL	Specified Acceptable Fuel Design Limit
SAR	Safety Analysis Report
SARRDL	Specified Acceptable Radionuclide Release Design Limit
SER	Safety Evaluation Report
SiC	Silicon Carbide
TAVA	Time-Average Volume-Average
TCR	Transformational Challenge Reactor
TEDE	Total Effective Dose Equivalent
THTR	Thorium High-Temperature Reactor
TP3	TRISO Particle Performance Program
TR	Topical Report
TRISO	Tristructural Isotropic
UCO	Uranium Oxycarbide (a mixture of UO_2, UC, and UC_2)
UIUC	University of Illinois at Urbana-Champaign
UQ	Uncertainty Quantification
USNC	Ultra Safe Nuclear Corporation
V&V	Verification and Validation
2-MGEM	Two-Modulator Generalized Ellipsometry Microscope

1. INTRODUCTION

The Micro Modular Reactor (MMR™) is being developed by Ultra Safe Nuclear (USNC) to be built and operated at the University of Illinois at Urbana - Champaign (UIUC) as a Class 104(c) utilization facility in accordance with 10 CFR 50.21(c) and licensed under 10 CFR 50.

MMR fuel is comprised of tristructural isotropic (TRISO) particles embedded in silicon carbide (SiC) Fully Ceramic Micro-Encapsulated (FCM®) pellets that are stacked in columns in solid hexagonal graphite blocks. The MMR plant life is set by its core operating life, which is designed to be approximately 20 years with no need for refueling.

1.1. Report Content and Structure

This report provides an overview of MMR reactor technology, the regulatory basis relevant to fuel qualification that must be satisfied for licensing the MMR, previous high-temperature gas-cooled reactor (HTGR) and TRISO fuel experience, FCM fuel design, manufacturing, quality control, fuel performance, and FCM fuel qualification methodology, structured as follows:

- Section 1.3 provides an overview of key design features of the MMR
- Section 1.4 summarizes the regulatory basis relevant to MMR fuel qualification
- Section 2 provides the design of FCM fuel in the context of the MMR design and operating envelope
- Section 3 provides information on FCM fuel performance and failure modes
- Section 4 provides information on FCM fuel manufacturing and quality control
- Section 5 presents the methodology for FCM fuel qualification
- Section 6 provides conclusions and limitations

1.2. Purpose and Regulatory Request

This Topical Report (TR) provides USNC's methodology for qualification of the MMR fuel design. This fuel will also be used for UIUC research reactor. UNSC's fuel qualification program provides reasonable assurance that the MMR FCM fuel design can operate with a low failure rate and a level of fission product release consistent with the design basis analysis. Results from the fuel qualification program will be submitted through a licensing application to the U.S. Nuclear Regulatory Commission (NRC). It is requested that the U.S. NRC review Sections 5.1, 5.2, 5.3, and 5.4 and the acceptance criteria listed in Section 5.6 of this report and approve the fuel qualification methodology described therein. Following NRC approval of this fuel qualification methodology TR, it is expected that meeting the acceptance criteria in Section 5.6 of the TR by a license applicant will qualify the fuel for use as part of its application.

1.3. Background

1.3.1. High-Temperature Gas-cooled Reactors

HTGRs share several design features:

- Inert helium coolant
- Graphite moderator
- Large core heat capacity
- Low core power density
- TRISO particles that retain their integrity at high temperatures
- High-temperature core materials (graphite)

These features together with proper core and plant design provide the ability for HTGRs to withstand design basis accidents (DBAs – e.g., extended loss of forced cooling) with minimal fission product release and minimal core damage. An HTGR design isolates most fission products within a fraction of a millimeter of where they were formed, simplifying plant design by containing and confining most radioactivity within the plant, thereby reducing the need for mitigating consequences of fuel damage and large-scale redistribution of radioactivity.

HTGRs use TRISO-coated particle fuel. Coated particles in early reactors consisted of a single layer of pyrocarbon (PyC). Fuel particle designs evolved rapidly in the 1960's to include a buffer layer to protect the single layer coating from damage, and eventually to variants of the current multi-layer TRISO particle to reduce fission product release. Improvements in TRISO manufacturing processes and quality control were made throughout the 1970's and 1980's, largely driven by the German Arbeitsgemeinschaft Versuchsreaktor (AVR) and Thorium High-Temperature Reactor (THTR) development programs.

Modern TRISO fuel plays a significant role in HTGR safety performance. TRISO fuel particles fabricated to tested specifications have low failure rates and maintain the ability to retain fission products at temperatures exceeding 1600 °C for several hundred hours.

1.3.2. Key Design Features of the MMR

The MMR is an HTGR designed to operate at 15 MW(t). For UIUC, power will not exceed the research reactor power limit. However, fuel qualification discussed in this TR applies generically to MMRs and therefore, is based on 15 MW(t).

The MMR uses an inert gas (helium) as the heat transfer fluid. The reactor will be fueled with EU (enriched uranium) at an enrichment from 5% to 19.75% ^{235}U in the form of TRISO particles embedded in SiC FCM pellets that are stacked in columns in solid hexagonal graphite blocks. The MMR has a plant life set by its core operating life, which is designed to be approximately 20 years with no need for refueling. The MMR is designed for passive safety response to design basis

accidents (DBAs) and relies on functional containment as the primary means to limit release of radioactivity to the environment.

As such, the MMR uses technology and safety capabilities considerably different from Light Water Reactor (LWR) technology that is the focus of many regulations. For example, the MMR does not require an active or passive emergency core cooling system (ECCS) to rapidly replenish primary coolant to recover the fuel in the event of a rupture of the primary pressure boundary. Large safety margins are provided by both the fuel and the reactor design.

- The fuel is comprised of TRISO particles, which provide a highly effective fission product retention capability. In response to an Electric Power Research Institute (EPRI) TR on the performance of TRISO fuel [1], the U.S. NRC issued a Safety Evaluation Report (SER) [2] [3] with some limitations and conditions that are considered in the MMR design. The superior fission product retention capability of TRISO fuel particles enables the concept of "functional containment" in which these particles serves as the first containment barrier when operated within the range of qualification parameters.
- The TRISO particles in MMR fuel are encased in an FCM pellet of SiC that provides an additional layer of defense-in-depth for the retention of fission products by functional containment.
- The low power density of active fuel region leads to slow fuel heat-up during loss of heat removal events.
- Low thermal power results in a small inventory available for release of the most limiting short-lived fission products for public safety, such as ^{131}I and ^{85}Kr. The increased inventory of long-lived fission products associated with a long core life is addressed by the defense-in-depth approach to functional containment.
- The low power rating also reduces the decay heat that must be removed in postulated accidents, simplifying passive decay heat removal.
- Heat transfer fluid used for core cooling during normal operation is an inert, chemically stable, single-phase gas (helium) at less than {{ }}
- Safety-related core cooling is passive and capable of maintaining fuel and component temperatures below limits with no helium, electrical power, or operator action.
- Secondary heat transfer is performed by a molten salt loop that effectively isolates the reactor from transients in the adjacent plant power conversion system.
- The reactor is below ground. Although it does not have nor need a leak-tested containment building, it is surrounded by a concrete structure (the citadel) that serves as a barrier to release of radioactivity to the environment and provides protection against external hazards.

Table 1.1 provides a high-level summary of key design features of the MMR and, in comparison, to current LWRs.

Functional containment is defined in Regulatory Guide (RG) 1.232 [4] as a barrier or set of barriers that effectively limit physical transport of radioactive material to the environment. The multiple layers in TRISO particles work together as a proven system to provide the first functional

containment barrier in the MMR design. {{

}} The FCM SiC pellet structure surrounding the TRISO particles provides an additional barrier to the release of fission products. {{

}}

The SiC used in FCM is resistant to oxidation for extended times at high temperatures [5]. The dissociation temperature of FCM SiC is well beyond MMR core temperatures during normal and accident conditions. Therefore, the design of FCM fuel adds another close-in, high-integrity layer to the existing multiple barriers in the TRISO fuel particles to enhance functional containment of radioactive fission products. The FCM fuel is designed to limit dose consequence during maximum hypothetical accidents or DBAs.

The primary fuel qualification objectives for the MMR are to demonstrate that the combination of standard TRISO particles embedded in a SiC matrix provides improved fission product retention. Specifically, the TRISO particles and the FCM pellet structure represent two separate and effective barriers designed to retain fission products during normal and accident conditions with a low failure rate and a level of fission product release consistent with the design basis analysis.

Table 1.1. *MMR Key Features and Differences from Operating LWRs*

Feature	MMR	LWR	Remarks
Operating power level	UIUC: lower of 15 MW(t) or allowed research reactor maximum power Power MMRs: 15 MW(t)	3000 to 4400 MW(t) (AP1000 3415 MW(t))	Full MMR power is less than decay heat of large LWR more than 24 hours after shutdown; short-lived fission product inventory is small
Heat transfer fluid	Helium – inert gas; single phase under all conditions; low stored energy	Water – also serves as moderator; scrubs fission products; high stored energy; undergoes phase change that causes high pressure and temperature in surrounding structure	Water coolant causes corrosion, and blowdown can damage safety systems by impingement, pressure, moisture, and temperature
Containment	Functional: TRISO integrity at high temperatures, with supplemental passive barriers for defense-in-depth, continuously confirmed by radiological monitoring while operating	Large containment building: subject to high pressure and temperature; many penetrations requiring active isolation, periodic leak testing, and maintenance	Functional containment is a barrier or set of barriers that effectively limit physical transport of radioactive material to the environment and serve as basis for the revised principal design criteria (PDC) in RG 1.232 [4]
Confinement	Citadel features	N/A	Below ground vault provides fission product barrier, shielding, protection from external hazards
Safety-related alternate current (AC) power systems	None	Class 1E AC distribution and emergency diesel generators	MMR safety is provided by passive systems
Refueling frequency	None (core and plant life are the same)	Every 1.5 to 2 years	Eliminates used fuel handling and storage risk
Fuel form	Uranium oxycarbide (UCO) TRISO particles encased in FCM pellets in a hexagonal graphite fuel block	Uranium dioxide pellets encased in zirconium alloy tubes	Negligible fission product release from MMR during operation or accidents
Fuel (^{235}U) enrichment	High-assay low-enriched uranium (HALEU); from 5% to 19.75% ^{235}U	Low-enriched uranium (LEU); lower than 5% ^{235}U	Both are LEU; MMR higher enrichment provides for longer core life
Fuel damage temperature	> 3272 ˚F (1800 ˚C)	2200 ˚F (1204 ˚C)	Zirconium-water reactions start at about 1800 ˚F (982 °C) in LWRs
Emergency replenishment of coolant	None; fuel limits met for unmitigated primary system blowdown	ECCS needed	Must quickly recover LWR fuel with water if loss of coolant occurs
Hydrogen management	External/internal flooding might release hydrogen (graphite-water reaction)	Zirconium-water reaction produces hydrogen if clad exceeds 2200 ˚F	Acceptance criteria limit mass of LWR fuel clad reacted
Primary system corrosion mechanisms	While helium itself is non-corrosive, contaminants must be controlled to low levels to avoid degradation of graphite and other materials	Various types of stress corrosion cracking; boric acid corrosion (PWRs)	Helium is inert whereas hot water is corrosive unless water chemistry is carefully controlled

1.4. Regulatory Basis

This section summarizes regulations relative to fuel qualification that must be satisfied for licensing the MMR, which is a non-water-cooled design. As such, it does not need to meet regulations with entry conditions limiting applicability to LWRs. Also, as a research reactor, the UIUC MMR is not required to meet regulations for power reactors. A TR has been developed to evaluate the applicability of regulations [6]. The fuel qualification activities described in this TR will be applicable to MMRs, including the non-power UIUC research reactor.

As the first MMR project in the United States, the UIUC application to the U.S. NRC will be for a construction permit including a Preliminary Safety Analysis Report (PSAR) in accordance with NUREG-1537 [7]. The UIUC PSAR will describe completed, on-going, and planned fuel qualification work that provides confidence in the acceptability of the design. The subsequent operating license application will include a final safety analysis report (FSAR) that provides justification for full power operation for several years. Following completion of accelerated long-life fuel testing, the data supporting full design life of the core will be provided to the U.S. NRC.

1.4.1. Regulations Relevant to the MMR Fuel Qualification

Current U.S. NRC regulations contain few requirements and limited information that is directly pertinent to the general process of – or specific requirements for – non-LWR fuel qualification. Instead, the regulations generally pertain to in-reactor fuel performance. The U.S. NRC has been focused on the preparation of 10 CFR Part 53 and has not included any detailed advanced, non-LWR fuel qualification guidance in the existing regulations.

As described in the UIUC TR on "Applicability of Nuclear Regulatory Commission Regulations" [6], regulations also may not be applicable to the UIUC MMR because of its designation as a Class 104(c) research reactor. This TR also addresses regulations potentially applicable to FCM fuel in future commercial MMRs. Although some of the regulations discussed in this section may not need to be applied to a research reactor, implementing them for fuel qualification will maintain consistency and provide appropriate assurance that UIUC MMR fuel will be satisfactory for use. The regulations addressed in this TR are:

- 10 CFR 20, "Standards for Protection against Radiation" – additional details are provided in the next section.
- 10 CFR 50 Appendix A, "General Design Criteria" – these criteria have been adapted in RG 1.232 [4] for advanced, non-LWRs. As an HTGR, the UIUC MMR will apply the criteria identified in Appendix C of RG 1.232, as discussed in the next section.
- 10 CFR 50.34, "Contents of applications: technical information" – this is a very broad section that describes the technical information to be included in a Part 50 (two-step licensing process) application and contains technical requirements that a reactor design must meet. Although many individual regulations have entry conditions that limit

applicability to water-cooled reactors, the content in 50.34 pertaining to fuel is primarily about fuel-clad metal water reactions, which is an LWR phenomenon.

- 10 CFR 50.34a, "Design objectives for equipment to control releases of radioactive material in effluents – nuclear power reactors" – although not applicable to the UIUC MMR, which is being licensed as a Class 104(c) research reactor, this regulation would be applicable to FCM fuel used in commercial power MMRs and is, consequently, addressed in this TR.
- 10 CFR 50.43, "Additional standards and provisions affecting Class 103 licenses and certifications for commercial power" – for HTGR power reactors, 10 CFR 50.43(e) requires that "The performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience, or a combination thereof." Although not specifically applicable to a Class 104(c) licensed research reactor, this item is considered technically relevant to fuel qualification. For fuel, the objectives of 50.43(e) will be satisfied by the FCM fuel qualification program described in this TR.
- 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors" – this is not applicable to the UIUC MMR. This regulation tends to dominate accident analysis for LWRs, but its closest equivalent (loss of helium heat transfer fluid) is not so limiting for an HTGR and can be accommodated with minimal fission product release without complex engineered safety features.

10 CFR Part 20

This part establishes radiation control requirements. In particular, 10 CFR 20.1301 establishes public and occupational radiation exposure limits for normal operations. In addition, a research reactor is required to meet the annual public dose limit for reactor accidents, as opposed to the 10 CFR 100 limits of 25 rem total effective dose equivalent [6].

10 CFR 50.34

50.34(a)(1)(ii) is not applicable to non-power reactors. To ensure no gaps are created by this exclusion, 50.34(a)(1)(ii)(D), as well as 10 CFR 52.47(a)(2)(iv) and 10 CFR 52.79(a)(1)(vi), were evaluated. They require that an applicant assume a fission product release from the core into the containment and that the applicant evaluate and analyze the postulated fission product release, using the expected demonstrable containment leak rate and any fission product cleanup systems intended to mitigate the consequences of the accidents. This language was deemed LWR-centric and not consistent with Staff Requirements Memorandum SECY-18-0096 [8] and RG 1.232 [4], which would allow functional containment for fission project retention rather than assuming a traditional pressure retaining containment [9]. Development of PDC for MMR by applying the guidance in RG 1.232 is considered to minimize potential gaps from the non-applicability of these specific LWR-based regulations.

The intent of the requirements in 10 CFR 50.34(a)(1)(ii)(D) will be met in the development of the MMR source term and radiological release modeling for DBAs.

10 CFR 50.43

According to 10 CFR 50.43(e), applications that propose nuclear reactor designs that differ significantly from LWR designs licensed before 1997 will be approved only if:

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

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

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

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

This TR describes actions completed and planned to ensure fuel performance has been adequately demonstrated through a combination of analysis, testing, and experience and availability of data to qualify analytical tools used for plant safety analysis.

1.4.2. Design Criteria

Current regulations pertinent to in-reactor fuel performance during normal operation, including anticipated operational occurrences (AOOs), are included in the General Design Criteria (GDC) contained in Appendix A to 10 CFR Part 50. As these GDCs were developed consistent with LWR experience, the U.S. NRC and the U.S. Department of Energy (DOE) engaged in a cooperative program to develop elements for an advanced reactor licensing framework. The result of public feedback and further review was the publication of RG 1.232 issued in April 2018 [4]. RG 1.232 provides guidance for adapting the LWR GDC for advanced reactors. 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) and Modular High-Temperature Gas-cooled Reactor (MHTGR) design criteria (MHTGR-DC).

- MHTGR-DC 10 discusses the specified acceptable system radionuclide release design limits (SARRDL) for radionuclide inventory released under normal and AOO conditions. The FCM pellets and TRISO particles comprise a series of fission product barriers and are expected to have a very low incremental fission product release during AOOs. The SARRDLs can be established so that the most limiting licensing basis event (LBE) does not exceed the regulatory dose criteria at the exclusion area boundary (EAB) and low-population zone. For the UIUC research reactor, these criteria are based on 10 CFR 20 (e.g., 10 CFR 20.1301 annualized dose limits to the public are not exceeded at the EAB for normal operation and AOOs), whereas the criteria of 10 CFR 100 apply to MMRs deployed as nuclear power plants. However, it is recognized that the concept of replacing specified acceptable fuel design limits (SAFDLs) with SARRDLs has not been approved by the U.S. NRC [4]. Therefore, the acceptance criteria in **Table 5.2** and discussed in Section 5.6 were developed to ensure fuel performance under design basis conditions satisfies goals for protection of public health and safety.

- 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: "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." This concept is based on the fact that the design relies upon high-integrity fuel particles encapsulated in pellets that provide an additional SiC barrier to minimize radionuclide release, and on a below-grade, safety-related concrete reactor building to retain and contain any radioactive releases, and to protect against external hazards [4].

- MHTGR-DC 26, Combined Reactivity Control Systems Capability, requires, in part, the ability to achieve and maintain safe shutdown under postulated accident conditions and provide assurance that the capability to cool the core is maintained.

1.4.3. NUREG-1537

For the UIUC MMR, NUREG-1537 [7] provides guidance on the conduct of licensing action reviews to U.S. NRC staff who review non-power reactor licensing applications and is the applicable standard review plan for the MMR. Chapter 4, in particular, provides guidance for including design features to ensure that the reactor can be safely operated and shut down from any operating condition or accident assumed in the safety analysis. After a summary description, the reactor core section of the safety analysis report (SAR) should contain the design information of all components of the reactor core, including reactor fuel. For each core component, the SAR should include design basis, system or component description, operational analyses and safety considerations, instrumentation and control features, and technical specifications requirements.

The reactor fuel section should include a reference to the fuel development program and the operational and limiting characteristics of the specific fuel used. The reviewer should be able to conclude that the applicant has included all information necessary to establish the limiting characteristics beyond which fuel integrity could be lost. Acceptance criteria for the information on reactor fuel are also provided in this reference.

Information to address these regulatory expectations is provided in the TR.

1.4.4. NUREG-0800

Although it is the standard review plan for LWRs, NUREG-0800 [10] provides useful perspective on fuel in its Section 4.2, Fuel System Design.

The following performance objectives should be met for LWR fuel:

1. The fuel system is not damaged (i.e., SAFDLs not exceeded) as a result of normal operation and AOOs,
2. Fuel damage will not prevent control rod insertion when required,
3. The number of fuel rod failures (i.e., amount of fission product release) is not underestimated for postulated accidents, and
4. Coolability (i.e., basic geometry and/or design basis heat removal path) is always maintained.

Section 4.2 of NUREG-0800 describes specific fuel damage criteria, most of which (e.g., stress/strain of metal components, fretting wear, chemical degradation such as oxidation and hydriding) are not relevant to HTGR fuel. Lessons learned from the development of the acceptance criteria in NUREG-0800 are incorporated in NUREG-2246 for advanced reactors [11].

1.4.5. NUREG-2246

The U.S. NRC states that the considerable experience base for traditional LWR fuel may not apply to proposed advanced reactor technologies because of differences in fuel designs and operating environments. Because fuel qualification is a lengthy process, the U.S. NRC staff began developing guidance in early 2020 on a performance-based fuel qualification approach for advanced reactors, including both power and non-power reactors. After obtaining and addressing public and Advisory Committee on Reactor Safeguards comments, the U.S. NRC issued NUREG-2246 [11].

NUREG-2246 provides a fuel qualification assessment framework (FQAF) for advanced reactor designs to support regulatory findings that nuclear fuel is qualified for use. In this framework, fuel manufacturing specifications and fuel safety criteria are systematically identified to demonstrate fuel qualification. For fuel manufacturing specifications, NUREG-2246 states

licensing documentation should include sufficient information to demonstrate the control of key parameters affecting fuel characteristics during the manufacturing process, as they can affect the performance during operation and accident conditions. This information includes dimensions and tolerances, constituents, and end-state attributes (e.g., microstructure). This TR provides relevant information in Section 4.2 and Section 4.3.

The safety criteria are generally associated with protection against the release of radioactive material but also address the fundamental safety functions of heat removal and reactivity control. Specifically, nuclear fuel is expected to retain its integrity under conditions of normal operation, including the effects of AOOs, although some degree of fuel failure can be accommodated for low-frequency DBA conditions.

In Appendix A of NUREG-2246, Table A-1 lists goals for the FQAF. The highest tier goals are summarized in **Table 1.2**. NUREG-2246 Appendix A also lists goals for the evaluation model and experimental data.

Table 1.2. *Top-level Goals for the Final Qualification Assessment Framework [11]*

Goal ID	Description
Top Level Goal	*Fuel is qualified for use*
G1	Fuel is manufactured in accordance with a specification
G1.1	Key dimensions and tolerances of fuel components are specified
G1.2	Key constituents are specified with allowance for impurities
G1.3	End-state attributes for materials within fuel components are specified or otherwise justified
G2	Margin to safety limits can be demonstrated
G2.1	End-state attributes for materials within fuel components are specified or otherwise justified
G2.2	Margin to radionuclide release limits under accident conditions can be demonstrated
G2.3	Ability to achieve and maintain safe shutdown is assured

1.4.6. U.S. NRC Policy Statements

No U.S. NRC policy statements directly apply to TRISO-coated particle fuel or to testing or monitoring of the fuel, nor does the U.S. NRC policy statement on the regulation of advanced nuclear power plants explicitly address fuel. However, many U.S. NRC activities associated with policy pertaining to source term have been identified over the years. Therefore, U.S. NRC expectations for fuel qualification for MMR fuel focus on demonstrating that the level of knowledge of source term behavior is sufficient.

Of the ten issues identified in SECY-93-092 [12], both “Containment Performance” and “Source Term” policy issues are related to TRISO fuel. The use of a multi-barrier containment configuration and associated mechanistic source terms for accident analyses are based on the performance of the TRISO fuel being both excellent and predictable [1]. More information is provided in the sections below.

Enclosure 1 of SECY-22-0008 [13] summarizes activities underway and planned by the U.S. NRC staff to support advanced nuclear technologies. This SECY covers the progress made in 2021 in six strategic areas: (1) staff development and knowledge management, (2) analytical tools, (3) regulatory framework, (4) consensus codes and standards, (5) resolution of policy issues, and (6) communications. Among the activities mentioned in this document, a series of reports documenting a comprehensive plan for developing computer code capabilities to support non-LWR reviews is cited. "Non-light Water (Non-LWRs) Reactor Vision and Strategy, Volume 2—Fuel Performance Analysis for Non-LWRs" [14] focuses on computer code readiness for fuel performance analysis. It discusses the high-level physics and phenomena that may need to be captured by a thermal-mechanical nuclear fuel performance code to support licensing reviews for non-LWR fuel designs, and the code development activities needed to adequately capture that physics.

Functional Containment Performance

The current LWR containment leakage requirements are provided in GDC 16 and Appendix J of 10 CFR Part 50. For advanced reactors that have operating conditions, coolants, and fuel forms significantly different than LWRs, different approaches to fulfilling the safety function of limiting the release of radioactive materials might be needed. This has led to the definition of "functional containment": a barrier or set of barriers that, taken together, effectively limit the physical transport of radioactive material to the environment [8].

In action on SECY-93-092 [12], the Commission approved the use of a standard based upon containment functional performance to evaluate the acceptability of proposed designs rather than rely on prescriptive containment design criteria. Functional containment should be assessed to show that:

- On-site and off-site radionuclide release limits are met
- For a period of approximately 24 hours following the onset of core damage, the specified containment challenge event results in no greater than the limiting containment leak rate used in evaluation of the event categories. After this period, the containment must prevent uncontrolled releases of radioactivity.

NUREG-1338 [15] observes that "If the overall safety of a plant design is improved (i.e., smaller accident dose consequences outside the containment) by reducing the requirements on the containment and increasing the integrity of fuel on an advanced reactor design, then there is an incentive to improve the fuel and there is a basis for accepting a different containment design."

In SECY-03-0047 [16], SECY-04-0103 [17], and SECY-05-006 [18], the U.S. NRC approved the use of a standard based on functional containment performance to evaluate the acceptability of the proposed designs, rather than relying on prescriptive containment design criteria. As part of the containment evaluation, the U.S. NRC instructed the staff to address the failure of the fuel particles, among other issues.

SECY-18-0096 [8] provides a methodology that can be used by non-LWR designers to define functional containment performance criteria in a manner that is technology inclusive, risk informed, and performance based. This document was approved by the Commission on December 4, 2018.

Source Term

The radiological source term for the MMR is defined as the quantities, timing, physical and chemical forms, and thermal energy of radionuclides (RNs) released from the reactor building to the environment during certain postulated LBEs. Because of the slow rate of RN release during accidents, the re-distribution during DBEs of previously deposited RNs is an important contributor. Therefore, fuel fabrication specifications must establish characteristics that provide high retention during both normal and accident conditions. These fuel specifications will allow establishing fuel performance limits (as SAFDLs and/or SARRDLs) that ensure minimal release of radioactivity for all routine and transient conditions.

In SECY-93-092 [12], the staff recommended source term calculations be based upon a mechanistic analysis for advanced reactors such as the Power Reactor Innovative Small Module, the MHTGR, and the Process Inherent Ultimate Safety reactor, provided that:

- The performance of the fuel is sufficiently well understood under normal and off-normal conditions to permit a mechanistic analysis
- Transport of fission products can be adequately modeled
- Bounding events are considered when developing the source terms

A mechanistic source term was described as:

> "…the result of an analysis of fission product release based on the amount of cladding damage, fuel damage, and core damage resulting from the specific accident sequences being evaluated. It is developed using best-estimate phenomenological models of the transport of the fission products from the fuel through the reactor coolant system, through all holdup volumes and barriers, taking into account mitigation features, and finally, into the environs."

In SECY-03-0047 [16], the Commission conditionally approved the staff's position that the source terms should be based on a mechanistic analysis. One of the conditions was that the database of characteristics and behavior of specific fuels be expanded, and the performance of the reactor and fuel under normal and off-normal conditions is sufficiently understood to permit a mechanistic analysis. There are several approaches that can be used to address the concern about sufficiency of fuel performance data:

- Obtain additional data specific to the fuel design to assess performance
- Restrict fuel characteristics to ranges where behavior under normal and accident conditions is well characterized

- Provide improved fission product retention capability through physical changes
- Perform source term analyses on a conservative basis to provide margin for uncertainties

For the MMR, all of the above are considered to some extent, as a defense-in-depth consideration. For example, USNC is performing its own fuel testing, fuel particle parameters are being kept close to those that have been used for the U.S. DOE TRISO fuel testing program (Section 1.5.3), FCM has replaced graphite as the TRISO particle matrix material, and conservative source term analysis is planned.

NUREG-1338

In 1989, NUREG-1338 [15] documented the U.S. NRC staff's review of the MHTGR design and its conclusions from the review as a draft preapplication safety evaluation report (PSER). A draft of the final PSER, which included U.S. DOE additional information and the content from two meetings held by the U.S. NRC, was completed in December 1995 [19]. The draft final PSER states that the information provided for the MHTGR up to that time had not demonstrated the necessary design and quality of fuel to adequately meet the performance objectives, as a number of parameters still needed to be defined. These parameters included quality control of the manufacturing process for the fuel and resulting tolerances on the coatings, expected fuel temperature, and potential dose consequences. Additionally, NUREG-1338 indicated that a statistically significant demonstration of the following was still missing:

1. The reference fuel manufacturing processes and quality-control methods ensure the production of fuel that meets specification requirements
2. The fuel fabricated using the reference fuel manufacturing processes meets the fuel performance requirements under normal operation and all credible accident conditions
3. Validated methods are available to accurately predict fuel performance and fission-product transport.

The information provided in this TR addresses completed, in progress, and planned activities that address these gaps.

Other Guidance

ANSI/ANS-53.1 [20], a standard developed by the American National Standards Institute (ANSI) and the American Nuclear Society (ANS), defines the process for specifying criteria to assure that Modular Helium Reactor (MHR) plants are designed so that they can be constructed and operated safely without undue risk to public health and safety. This purpose is achieved through the identification of applicable safety requirements from the national nuclear regulator, industrial codes and standards, and other published guidance and professional engineering practices.

1.5. Previous Experience with Qualification of Coated Particle Fuel

1.5.1. Peach Bottom

An HTGR construction permit was issued to Philadelphia Electric Company for the Peach Bottom Unit I plant in 1962. This 40 MW(e) plant operated from 1967 to 1974. Reference [21] states that the Peach Bottom plant Class 104 operating license was granted based in part on the final hazards summary report issued by Philadelphia Electric Company. The Peach Bottom final hazards summary report used a conservative source term based on mechanistic release phenomena and preserved the time-dependent nature of HTGR fuel release during a beyond design basis accident (BDBA). The initial fuel used a bistructural isotropic (BISO) particle that had only a single pyrolytic carbon (PyC) layer surrounding the uranium carbide kernel. The second core in 1970 added a buffer layer between the kernel and PyC coating. The fuel was contained in graphite blocks.

1.5.2. Fort St. Vrain

The Fort St. Vrain (FSV) Nuclear Generating Station was a prismatic fuel HTGR that generated 842 MW(t) to achieve a net output of 330 MW(e). FSV was operated from 1974 to 1989 and was licensed using a deterministic source term based on the technical information document TID-14844 [22]. The FSV fuel was highly enriched uranium/thorium carbide fissile and thorium carbide fertile TRISO particles. Two DBAs were presented in the FSV FSAR [23]: loss of forced circulation and rapid depressurization of the reactor vessel. The maximum credible accident was selected as failure of the regeneration line. The U.S. NRC staff assumed an unfiltered release about 70 times more and atmospheric dilution factor about 30 times less than that of the applicant's analysis. These calculations represented an early effort to use a mechanistic source term approach for a medium-sized HTGR. The applicant did not consider TID-14844 values to be applicable to the HTGR system, but a mechanistic source term was compared to TID-14844 assumptions to demonstrate the relative safety of the HTGR.

1.5.3. AGR Program

The U.S. DOE initiated the Advanced Gas Reactor (AGR) Fuel Development and Qualification Program in 2002 to establish U.S. capability to fabricate high-quality UCO TRISO fuel and demonstrate its performance [24].

The first two fuel irradiation tests in the program, designated AGR-1 and AGR-2, assessed UCO fuel performance during irradiation and in post-irradiation high-temperature accident safety tests. A high level of fission product retention was achieved within the bounds of the irradiation test [25] [26].

AGR-3/4 irradiation experiments tested UCO fuel compacts that contained designed-to-fail fuel particles to provide irradiation performance and fission product transport and release data.

Capsule sweep gas monitoring with a gamma-ray spectrometer and post-irradiation examination (PIE) were used to obtain quantitative data [27].

AGR-5/6 irradiation experiments were performed to verify successful performance of the reference-design fuel by demonstrating compliance with statistical performance requirements under normal operating conditions [28].

The AGR-7 test was designed to explore fuel performance at higher fuel temperatures with the primary objective to demonstrate the capability of the fuel to withstand conditions beyond normal operating conditions in support of plant design and licensing [28].

The AGR test experience provides confidence that TRISO particle failure fractions in particles manufactured to the same specification in a quality-controlled program will result in similar very low failure fractions under similar irradiation conditions. The MMR fuel particle design and specification are based on the AGR fuel particle specifications for all critical parameters, with the expectation that they will behave consistently with the AGR irradiation tests.

1.5.4. Next Generation Nuclear Plant

In 2005, the U.S. DOE established the Next Generation Nuclear Plant (NGNP) project at Idaho National Laboratory to support near-term commercial deployment of an 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 the U.S. NRC staff and focused on adapting existing nuclear power plant regulatory requirements to the needs of NGNP licensing [29].

Within the NGNP program, Battelle Energy Alliance subcontracted with three industrial teams for engineering studies in support of NGNP technology development and licensing. As part of the contractual work scope, a functional analysis methodology was used to establish a defensible basis for the in-core fuel performance criteria and the as-manufactured fuel quality specifications for a prismatic MHR with a 750 °C core outlet temperature [30].

1.5.5. TRISO EPRI Topical Report

In 2020, EPRI published *Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO)-Coated Particle Fuel Performance: Topical Report, EPRI-AR-1(NP)* [1] to document key data and results from the first two campaigns of the AGR Program (AGR-1 and AGR-2). The report provides the technical bases (that is, particle design, irradiation, and accident testing results) that demonstrate the functional performance of UCO TRISO particles.[1]

The EPRI report was submitted to the U.S. NRC as a TR for formal review and issuance of a SER [2]. The SER concluded that there is reasonable assurance that TRISO particles produced to the

[1] The AGR-2 irradiation test also included UO_2 fuel, but only UCO fuel performance is addressed in the EPRI TR.

specifications and limited to the performance parameters documented in the TRISO TR will satisfy a portion of the requirements associated with PDC 10, subject to the Limitations and Conditions in Section 4.0 of the SER. More specifically, TRISO particles produced to the specifications within the TR and limited to the performance parameters in the TR will perform in accordance with the AGR data presented in Sections 6 and 7 of the TR. Therefore, the data can be used to support safety analyses referencing the unique design features of TRISO particles.

2. DESIGN OF FCM FUEL

2.1. Development of Coated Particle Fuels

A historical overview of coated particle fuel is provided in [31]. Significant progress on coated particle fuels began with the Dragon project [32] in the United Kingdom in an effort to control fission product release using single layers of PyC applied directly onto the fuel kernel using a tumbling process.

In parallel, work on HTGR fuel in the United States, Austria, Belgium, France, Germany, and the United Kingdom led to the development of improved manufacturing and quality control processes. Additional features such as low density PyC layers and SiC layers were incorporated into the design. Both the Peach Bottom and Dragon reactors were used as test beds for the development of coated particle fuels, with Peach Bottom using BISO as its standard fuel for the last four years of operation. The AVR began operation in the late 1960's using a BISO fuel variant and provided a test bed for coated particle fuel development for 21 years [33].

Extensive development of TRISO fuels and understanding of TRISO fuel manufacturing and performance occurred in the 1970's and was widely published [34], providing a technical basis for use in commercial reactors. In 1982, testing of fuel elements containing UO_2 TRISO began in AVR combining the state of knowledge of TRISO fuel performance with the application of rigorous quality control. Continuing for six years, this test campaign proved that TRISO fuel could be mass produced with low defect fractions and repeatable performance. The German fuel development program was conducted through the mid 1990's, resulting in a large amount of data on fabrication, irradiation, and PIE.

The FSV reactor, an 842 MW(t) prismatic core HTGR that used TRISO particles with fissile $(Th,U)C_2$ and fertile (ThC_2) kernels provided a large-scale commercial demonstration of TRISO fuel manufacturing and performance from 1981-1989 [35]. For FSV, 2448 hexagonal fuel elements, 7.1 million fuel compacts, and 26,600 kg of TRISO coated fuel particles were produced. Irradiation testing of FSV fuel in the F-30 capsule test was used to demonstrate that HTGR requirements for fuel quality and fission product release were acceptable [36]. Irradiated fuel temperatures greater than 1300 °C, maximum burnup of 16% fissions per initial heavy metal atom (FIMA), and a maximum fast neutron fluence of 4.5×10^{25} n/m^2 (E > 0.18 MeV) were achieved with no evidence of significant in-service coating failure. Measurement of circulating ^{85m}Kr and ^{138}Xe fission products during subsequent operation was lower than that expected based on testing and computer modeling [37]. Estimates of the release of condensable fission products from collectors after final reactor shutdown indicate that release of ^{90}Sr, ^{134}Cs, and ^{131}I was overpredicted by fuel performance and fission product release, transport, and plate-out models, while release of ^{137}Cs was slightly underpredicted. The FSV carbide kernels were found to be susceptible to water corrosion and kernel migration and to have poor retention of lanthanides, resulting in adoption of UCO kernels for the design of the Gas Turbine Modular Helium Reactor [37].

In 2002, the U.S. DOE initiated the AGR program to renew U.S. development of LEU UCO TRISO fuel. The objectives of the AGR program were to establish a domestic, commercial TRISO fuel fabrication capability in the USA and to generate fuel performance data to support the design, licensing, and operation of HTGRs in the USA under the umbrella of NGNP. Although the U.S. DOE discontinued the NGNP project in 2011, the AGR program continued its fuel development activities to support the licensing of high-temperature reactor designs by U.S. commercial reactor vendors. To accomplish these objectives, the AGR program manufactured high-quality TRISO fuel with low coating defect levels and successfully performed irradiation testing and post-irradiation safety testing of this TRISO fuel under normal reactor operating and anticipated accident conditions.

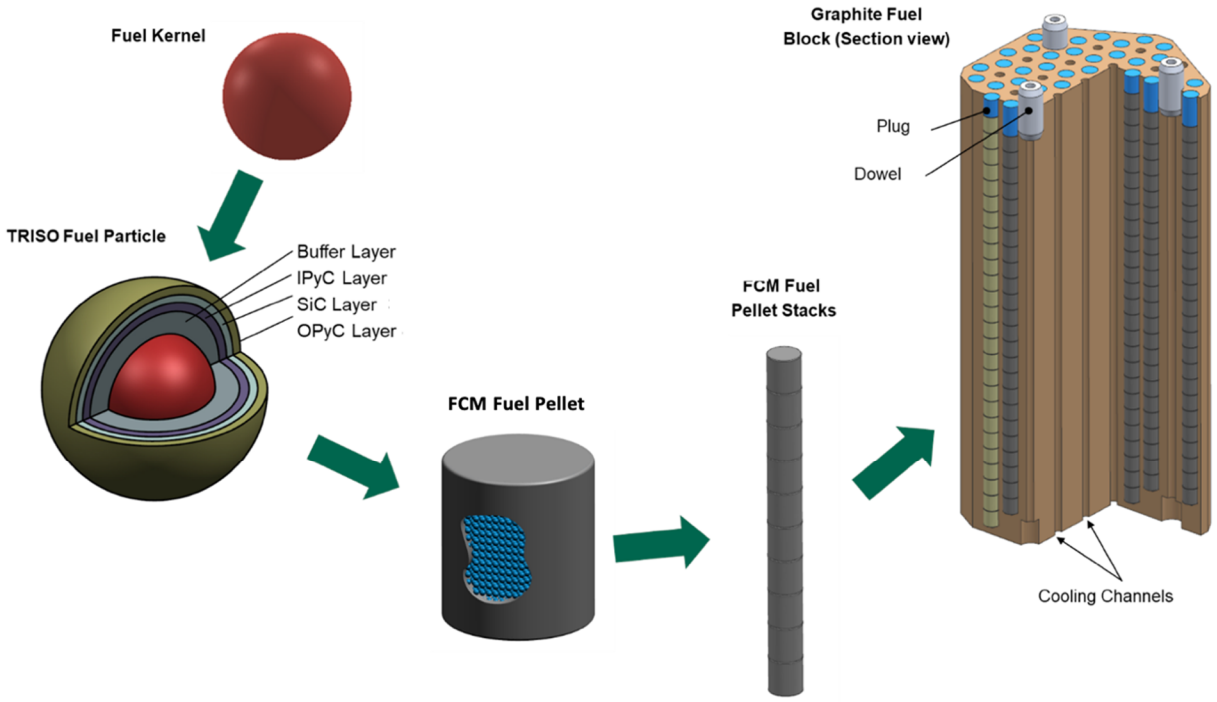
The AGR irradiation consisted of seven tests grouped into four different campaigns:

- AGR-1: This experiment was a shakedown test of the multi-capsule, instrumented test train design that would be used in all subsequent experiments; it was meant to assess the performance of TRISO-coated particles and compacts fabricated at the laboratory scale; the irradiation was performed from December 2006 to November 2009 and achieved a record peak burnup of 19.6 %FIMA [25].
- AGR-2: This experiment was a performance demonstration of pilot-scale TRISO-coated particles in lab-scale compacts; in addition to UCO, the test included UO_2 fuel to compare the performance of both kernel types and to satisfy the interest in UO_2-fueled pebble-bed reactors; the irradiation was performed from June 2010 to October 2013 and achieved a time-average peak temperature of 1360 °C [26].
- AGR-3/4: This experiment was aimed at studying fission product transport in fuel compact matrix material and reactor-grade graphite; the irradiation was performed from December 2011 to April 2014 [27].
- AGR-5/6/7: This experiment was the fuel qualification (AGR-5/6) and performance margin (AGR-7) irradiation experiment; the irradiation was performed from February 2018 to July 2020 and achieved a peak burnup of 15.3 %FIMA, peak fast neutron fluence of 5.55×10^{25} n/m^2 ($E > 0.18$ MeV), and peak time-average temperature of 1405 °C [28].

2.2. Description of FCM Fuel Design

USNC MMR fuel consists of TRISO fuel particles embedded in a ceramic matrix to form an FCM fuel pellet. Burnable absorber pellets are composed of mixtures of SiC and boron carbide (B_4C) sintered to low density. The fuel and burnable absorber pellets are inserted into a graphite block that together comprise the fuel element. A schematic of the MMR fuel design is provided in **Figure 2.1**.

Figure 2.1. *MMR Fuel Design*



TRISO fuel particles are well developed and have known failure and fission product retention characteristics [1] [2] when manufactured to a well-defined specification, as provided in Table 5-5 of reference [1]. Systematic analyses were performed by the AGR program using the PARFUME code [38] to determine which fuel attributes are critical to fuel performance and to determine the appropriate critical limits for those attributes [39]. {{

}}

The 20-year life of the MMR core requires that the functional containment performance of the fuel be maintained during the core lifetime. Replacement of the traditional carbonaceous TRISO fuel matrix with a SiC matrix provides an additional barrier to fission product release. The TRISO UCO fuel kernel retains many fission products. Three fission product barrier layers outside of the fuel kernel, the inner PyC (IPyC), SiC, and outer PyC (OPyC) layers provide additional retention. The SiC layer is the most effective barrier to the release of radioactivity from TRISO particles. {{

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Figure 2.2 *Graphical Representation of FCM Fuel Architecture (not to scale)*

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The refractory SiC pellet matrix has excellent thermal and chemical stability which effectively precludes gross changes in fuel geometry and fission product release due to chemical attack. The effectiveness of SiC as a barrier to fission product release has been demonstrated and characterized through the extensive TRISO fuel testing conducted by the AGR program and other TRISO fuel development programs as described in Section 2.1. Irradiation testing of FCM SiC matrix material [40] [41] and FCM fuel [42] has been conducted by the U.S. DOE Transformational Challenge Reactor (TCR) program and is currently in progress.

{{

}} FCM pellets are manufactured using advanced manufacturing methods designed to embed the TRISO particles in the SiC matrix without damage to the particles. The fuel manufacturing process is discussed in Section 4.

{{ }} The major design parameters for the graphite blocks are provided in **Table 2.1** and a graphical description is shown in **Figure 2.3**. The functions of the graphite block are to maintain the geometry of the fuel and fixed burnable absorber pellets, to transfer heat from the fuel to the coolant, to moderate neutrons, to channel the helium coolant, and to maintain a coolable and controllable fuel geometry.

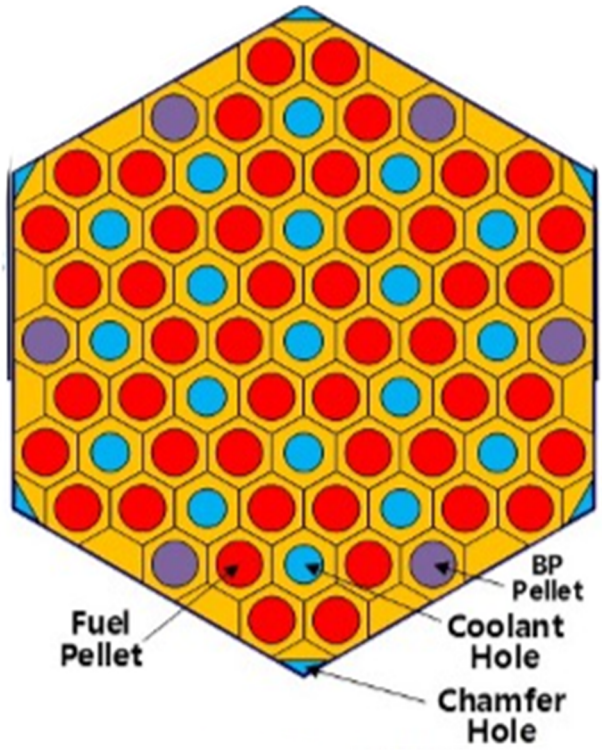
{{

}} At each element-to-element interface in a column, there is a dowel/socket connection which aligns rod and coolant channels, and also transfers seismic loads on fuel element.

Table 2.1. *Major Design Parameters of the Graphite Fuel Block*

Property	Fuel Block
Flat-to-Flat length (mm)	{{ }}
Height (mm)	{{ }}
Number of fuel holes	{{ }}
Fuel hole diameter (mm)	{{ }}
Gap with fuel pellet (mm)	{{ }}
Number of cooling channels	{{ }}
Number of chamfer channels	{{ }}
Coolant hole diameter (mm)	{{ }}
Number of burnable poison holes	{{ }}

Figure2.3. *Graphical Description of an MMR Graphite Block*



2.2.1 FCM TRISO Coated Particle Fuel Design

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}} Nominal values from the preliminary fuel product specification for FCM TRISO are compared to the AGR-1 and AGR-2 TRISO fuel specifications in **Table 2.2**.

UCO kernel chemistry (a mixture of UO_2, UC, and UC_2) improves on historical UO_2 TRISO fuel design by limiting production of CO through the addition of carbon. Oxygen released from the kernel during fission reacts with the UC_x phases to form uranium oxide, preventing the formation of CO by reaction with the carbon layers surrounding the kernel. Mitigating CO production reduces risk of kernel migration, over-pressurization, and chemical attack on the SiC layer.

The porous carbon buffer layer, located between the kernel and IPyC layer, mechanically isolates kernel swelling from the IPyC layer, absorbs fission fragments, and provides a void volume to accommodate fission gases and limit buildup of internal pressure. The SiC layer acts both as main load-bearing structural layer against internal fission gas pressure and as main barrier to metallic fission product release. It is sandwiched between the two dense PyC layers. The IPyC and OPyC layers help to maintain the mechanical integrity of the TRISO particle and provide effective barriers to fission gas release.

Table 2.2. *Comparison between USNC and AGR-1 and AGR-2 TRISO Specifications [25] [26]*

Particle Property	{{ }}	AGR-1 TRISO Baseline	AGR-2 UCO TRISO	AGR-2 UO_2
	{{ }}	Mean Measured Value	Mean Measured Value	Mean Measured Value
Kernel Type	{{ }}	UCO	UCO	UO_2
Kernel Diameter (µm)	{{ }}	349.7	426.7	507.7
Kernel Density (g/cm^3)	{{ }}	10.92	10.97	10.86
Kernel C/U (molar ratio)	{{ }}	0.325	0.392	-
Kernel O/U (molar ratio)	{{ }}	1.361	1.428	2.003
Kernel [C+O]/U (molar ratio)	{{ }}	1.685	1.818	2.003
Buffer Thickness (µm)	{{ }}	103.5	98.9	97.7
IPyC Thickness (µm)	{{ }}	39.4	40.4	41.9
SiC Thickness (µm)	{{ }}	35.3	35.2	37.5
OPyC Thickness (µm)	{{ }}	41.0	43.4	45.6
Buffer Density (g/cm^3)	{{ }}	1.10	~1.04	0.99
IPyC Density (g/cm^3)	{{ }}	1.904	1.89	~1.89
SiC Density (g/cm^3)	{{ }}	3.208	3.197	3.200
OPyC Density (g/cm^3)	{{ }}	1.907	1.907	1.884
IPyC Anisotropy (BAF_{True})	{{ }}	1.015	1.024	1.023
OPyC Anisotropy (BAF_{True})	{{ }}	1.013	1.018	1.015
SiC Sphericity/Aspect Ratio	{{ }}	-	1.037	1.034
OPyC Sphericity/Aspect Ratio	{{ }}	1.054	1.052	1.052
Particle Diameter (µm)	{{ }}	-	-	-
IPyC Defect Fraction	{{ }}	≤ 6.1×10^{-5}	≤ 4.8×10^{-5}	≤ 7.7×10^{-5}
SiC Defect Fraction	{{ }}	≤ 1.3×10^{-4}	≤ 1.2×10^{-5}	≤ 2.5×10^{-5}
Missing OPyC (Defect Fraction)	{{ }}	≤ 6.1×10^{-5}	≤ 9.5×10^{-4}	≤ 2.0×10^{-3}
Exposed Kernel Defect Fraction	{{ }}	≤ 3.1×10^{-5}	Not reported	Not reported
Uranium contamination fraction (dispersed uranium + exposed kernel fraction)	{{ }}	-	≤ 2.5×10^{-5}	≤ 3.2×10^{-5}
Dispersed uranium fraction (U contamination fraction excluding exposed kernels)	{{ }}	3.64×10^{-7}	3.94×10^{-6}	9.66×10^{-7}

{{

}}

Although the kernel chemistry differs from FCM UCO TRISO, data from AGR-2 UO_2 TRISO fuel is included in **Table 2.2** as an example of fuel {{ }} that exhibited good performance during irradiation to a peak burnup of 10.7 %FIMA [43]. UO_2 TRISO generates CO gas in addition to gaseous fission products [44], resulting in higher internal gas pressure relative to UCO.

2.2.2 FCM Fuel Pellet Design

The MMR FCM fuel pellet is a SiC cylinder that contains the FCM TRISO particles, shown in **Figure 2.4**. {{

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Figure 2.4. *Graphical Description of an FCM Fuel Pellet*

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The fuel pellet is produced using additive manufacturing (AM) using binder jet printing and chemical vapor infiltration (CVI) processes. Initially, a pre-formed cylindrical SiC shell with bottom and radial surfaces is printed with SiC feedstock powder. The low-density green shell is strengthened by partially infiltrating with methyltrichlorosilane (MTS) in a CVI system to create crystalline SiC bonds between the SiC particles and to allow for handling, inspection, and loading

of the shell. The strengthened shell is loaded with TRISO particles and SiC powder, assisted by light vibration during filling. The vibration does not result in any significant settling or mechanical forces on the TRISO particles or fuel pellet shell. After TRISO loading is completed, the upper region of the shell is filled with SiC powder to provide a substrate for deposition of CVI SiC.

The fuel pellet, now filled with TRISO particles and SiC powder and capped with SiC powder, is subjected to a second CVI process for final densification. {{

}} The preliminary specifications of the FCM fuel pellet are provided in **Table 2.3**.

Table 2.3. *Preliminary MMR FCM Fuel Pellet Properties*

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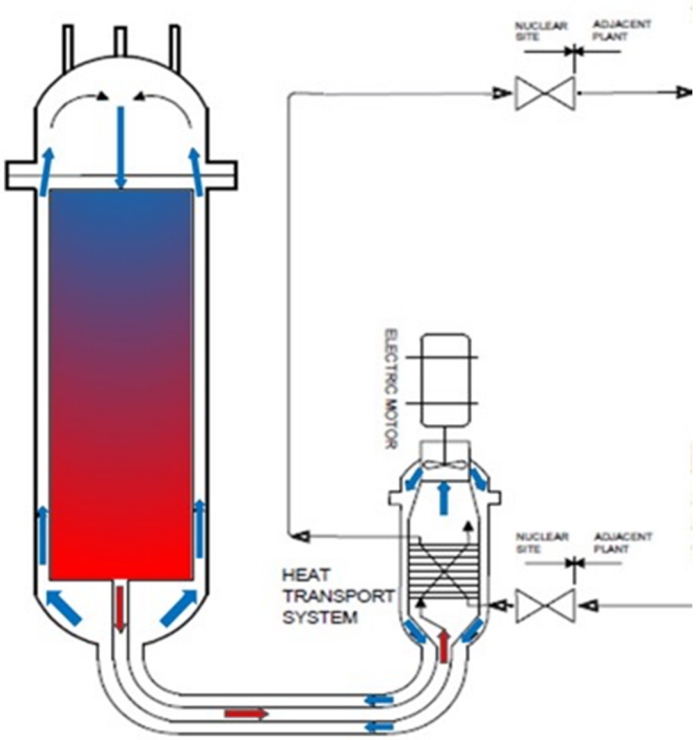
2.3. MMR Core Design

The MMR is an HTGR that is designed to operate at 15 MW(t), but the maximum power will be set at that permitted under a research reactor license. The MMR is planned to be licensed at UIUC as a Class 104(c) Non-power Production or Utilization Facility, in accordance with 10 CFR 50.21(c). The MMR provides heat to an independent, non-nuclear Adjacent Plant via an intermediate molten salt loop. The Adjacent Plant can use the heat from the Nuclear Plant to make steam for electricity generation, off-site process heat, or district heating.

The reactor core is graphite-moderated, helium-cooled, and fueled with FCM pellets. The FCM fuel system is designed to provide improved fission product retention, chemical and thermal stability, and strong negative reactivity feedback coefficients. These inherent safety characteristics effectively eliminate the need for an active safety-related ECCS to maintain fuel integrity and greatly limit the potential radioactivity release during accidents.

The MMR uses helium gas as the primary coolant. A circulator forces helium up along the outer annulus of the reactor vessel into an upper plenum where it turns and flows downward through the core coolant channels and eventually to the intermediate heat exchanger where heat is transferred to the molten salt loop. **Figure 2.5** illustrates the helium flow path. Helium is chemically and neutronically inert and eliminates adverse reactions between the coolant, graphite, burnable absorber, FCM fuel, and the primary coolant pressure boundary. Because helium remains gaseous under all normal and accident conditions, no flashing or boiling of the coolant is possible, pressure measurements are certain, no coolant level measurements are required, and pump cavitation cannot occur.

Figure 2.5. *MMR Helium Flow Path*



The two separate, independent, and diverse means of shutting down the reactor are:

1) The helium circulator trip is a fast-acting means of reactor shutdown. It quickly renders the reactor in a subcritical, hot shutdown state from normal operation, AOOs, and DBAs, using the inherent negative temperature reactivity feedback of the fuel (Doppler).
2) The Reactivity Control and Shutdown System (RCSS) rods trip is provided as a diverse means of reactor shutdown. It takes the reactor subcritical to the guaranteed shutdown state from normal operation, in AOOs, DBAs, and BDBAs and maintains subcriticality for the most limiting conditions of the reactor core, including hypothetical severe degradation of the core.

These means of shutdown are diverse, passive, separate, and independent from each other and can be initiated with an automatic trip or manual actuation.

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Figure 2.6. *Core Arrangement of RCSS Control Rods*

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The high core heat capacity, good thermal conductivity, and robust fuel design are capable of absorbing and dissipating enough decay heat to prevent general fuel damage or substantial fission product release for any DBE, including loss of electrical power to the helium circulator and the helium coolant inventory. These means of cooling are passive: they do not require electrical power or operator action. In the event of failure of normal shutdown cooling using the reactor cavity cooling system (RCCS), decay heat will be removed via conduction and thermal radiation through the core graphite, the reactor pressure vessel, reactor cavity, and RCCS to the ground surrounding the reactor cavity.

2.4. MMR Fuel Operating Envelope

FCM TRISO fuel particle properties are given in **Table 2.2**. {{

}} unless otherwise noted. The 15 MW(t) operating power bounds research reactor operating power limits. {{

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Preliminary MMR fuel operating parameters are provided in **Table 2.4** and **Table 2.5** for a reactor operating at 15 MW(t) power. **Table 2.4** provides core average conditions for 19 effective full power years (EFPY) of operation for comparison with peak values during normal operation. **Table 2.5** provides the maximum fuel values during normal operation during the 19 EFPY core lifetime.

Table 2.4. *Preliminary FCM Fuel Operating Conditions in the MMR*

Parameter	Value
Core lifetime (EFPY)	19
{{ }}	{{ }}
{{ }}	{{ }}
{{ }}	{{ }}
{{ }}	{{ }}

The MMR normal operating envelope is compared to AGR-1 and AGR-2 irradiation test parameters in **Table 2.6** and **Table 2.7**. {{

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The operating temperatures of FCM fuel pellets in the MMR core are graphically depicted in **Figure 2.7** relative to AGR-1 and AGR-2 test parameters and the decomposition temperature of SiC. The low power density MMR design does not challenge the temperature limits of TRISO particles or of the FCM matrix SiC.

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Table 2.5. *MMR Core Lifetime Fuel Operating Values*

Parameter	{{ }}	{{ }}
Maximum burnup (%FIMA)	{{ }}	{{ }}
Maximum pellet power (W)	{{ }}	{{ }}
Coated particle maximum power (mW)	{{ }}	{{ }}
Maximum fuel peak centerline temperature (° C)	{{ }}	{{ }}
Maximum fast neutron fluence (10^{25} n/m^2, E > 0.18 MeV, n/m^2)	{{ }}	{{ }}

Table 2.6. *Estimated MMR FCM Operating Burnup and Fast Neutron Fluence and Comparison with AGR-1 and AGR-2 Irradiation Test Conditions*

Capsule	Fuel Burnup (%FIMA)			Fast Neutron Fluence (10^{25} n/m^2, E > 0.18 MeV)		
	Minimum	Average	Peak	Minimum	Average	Peak
AGR-1 [(a)] UCO						
1	13.2	15.3	17.4	2.52	3.02	3.39
2	16.0	17.8	19.1	3.35	3.77	4.05
3	17.0	18.6	19.5	3.72	4.07	4.30
4	16.4	18.2	19.4	3.59	3.98	4.21
5	14.2	16.5	18.2	3.08	3.52	3.82
6	11.3	13.4	15.3	2.17	2.65	3.04
AGR-2 [(a)] UCO						
2	10.8	12.2	13.2	2.88	3.25	3.47
5	10.1	11.7	12.9	2.77	3.18	3.42
6	7.3	9.3	10.8	1.94	2.39	2.73
AGR-2 [(a)] UO_2						
3	9.0	10.1	10.7	3.05	3.35	3.53
{{ }}						
{{ }}	{{ }}	{{ }}	{{ }}			
{{ }}	{{ }}	{{ }}	{{ }}			

(a) Values for AGR-1 and AGR-2 are taken from Table 6-1 of reference [1].
(b) {{ }}

Table 2.7. *Estimated MMR FCM Operating Temperatures and Comparison with AGR-1 and AGR-2 Irradiation Test Conditions*

Capsule	Time-Average Minimum Temperature (°C)	Time-Average Volume-Averaged Temperature (°C)	Time-Average Maximum Temperature (°C)
AGR-1 [(a)] UCO			
1	854	1054	1167
2	800	1002	1124
3	828	1028	1147
4	866	1070	1187
5	818	1023	1144
6	885	1087	1197
AGR-2 [(a)] UCO			
2	1034	1252	1360
5	901	1101	1210
6	868	1074	1183
AGR-2 [(a)] UO_2			
3	889	1032	1105
{{		}}	
	{{		}}

(a) Values for AGR-1 and AGR-2 are taken from Table 6-4 of reference [1].
(b) {{ }}

Figure 2.7. *Schematic of MMR Fuel Temperatures*

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3. FCM FUEL PERFORMANCE

The MMR FCM fuel form is a cylindrical pellet composed of TRISO particles held in a SiC matrix and encapsulated in a high density SiC shell. {{

}} In particular, FCM fuel:

- Incorporates a TRISO particle design close to those of the AGR fuel that is demonstrated to have high fission product retention during tests exposing its particles to higher temperatures than expected for MMR operation, transients, and accidents.
- Surrounds the particles with a partially densified SiC matrix that protects TRISO particles from mechanical damage.
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The performance of TRISO particles with similar characteristics has been extensively tested and shown to provide effective retention of fission products at temperatures (> 1600 °C) and times (300 hours) that exceed postulated HTGR accident conditions. The testing and in-pile experience was obtained with TRISO particles embedded in carbon-based matrices. It follows that qualification of the FCM fuel design requires verification of the performance of TRISO particles embedded in the FCM fuel pellet, maintaining geometry, and providing an additional barrier to fission product release. TRISO coated particle fuel performance is discussed below.

3.1. TRISO Coated Particle Fuel Performance

TRISO particles consist of a fuel kernel surrounded by a porous buffer and three dense layers, namely a SiC layer sandwiched between IPyC and OPyC layers – referred to as outer coating layers.

As the fuel burnup in a particle increases, the kernel swells outward under the influence of solid and gaseous fission products. Swelling occurs because of the expansion of the kernel's atomic lattice and the formation of fission gas bubbles. Conversely, the buffer retracts inward away from the IPyC layer as it densifies and shrinks [45]. During the process, the kernel and buffer stay bonded, while the buffer tends to detach from the IPyC, creating a gap between the two layers.

The buffer-IPyC gap is the largest thermal resistance in the TRISO particle and largely determines the kernel temperature. The buffer constitutes the second main thermal resistance, though its thermal conductivity increases as it densifies throughout irradiation. Comparatively, the three outer coating layers have higher thermal conductivities and remain at temperatures very close to the temperature of the surrounding SiC matrix. As irradiation proceeds, the buffer layer is pushed outward by the swelling kernel and the width of the buffer-IPyC gap is determined by the balance between kernel expansion and buffer shrinkage.

The IPyC and OPyC layers exhibit shrinkage early in irradiation followed by swelling as fast neutron fluence increases. PyC in TRISO particles is slightly anisotropic and exhibits different irradiation-induced strain rates in its radial and tangential directions. PyC anisotropy is, however, limited by specification to values demonstrated to provide good overall TRISO performance. At low fast neutron fluence, PyC shrinks in both directions. As fast neutron fluence is accumulated, the radial strain changes from shrinkage to swelling. At even higher fast neutron fluence, swelling also starts occurring in the tangential direction. The reversing of the strain depends on the density and degree of anisotropy of the PyC layer (controlled by the TRISO specification) and on the irradiation temperature.

Because the elastic modulus of SiC is much higher than the (radial and tangential) elastic moduli of the PyC layers, the SiC layer acts as the primary structural layer in TRISO particles. Early in irradiation, PyC shrinkage creates tensile stress in the IPyC and OPyC layers, which imparts overall compressive stress onto the rigid SiC layer. The SiC layer has irradiation-induced dimensional changes that are negligible compared to PyC. Cracking of the PyC layers can occur if the tensile stress that results from shrinkage overcomes the fracture strength of the PyC layers. This can result in localized tensile stress on the SiC layer and potentially lead to SiC layer failure.

As irradiation progresses, irradiation-induced creep in the PyC layers offsets their shrinkage and relieves some of their tensile stress. Simultaneously, fission gas pressure builds up in the void volume of the particle internal to the IPyC layer. As internal pressure increases with burnup, the tangential stress in the SiC layer changes from compressive to tensile. This can potentially lead to the mechanical failure of the SiC layer if the tensile stress reaches a value that exceeds the strength of the particle's SiC layer. The irradiation behavior of the outer coating layers is shown schematically in **Figure 3.1** and summarized in **Table 3.1**. An illustrative representation of the corresponding stress in the IPyC and SiC layers as a function of fast neutron fluence is shown in **Figure 3.2**.

Figure 3.1. *Irradiation Behavior of Outer Coating Layers in a TRISO Particle [45]*

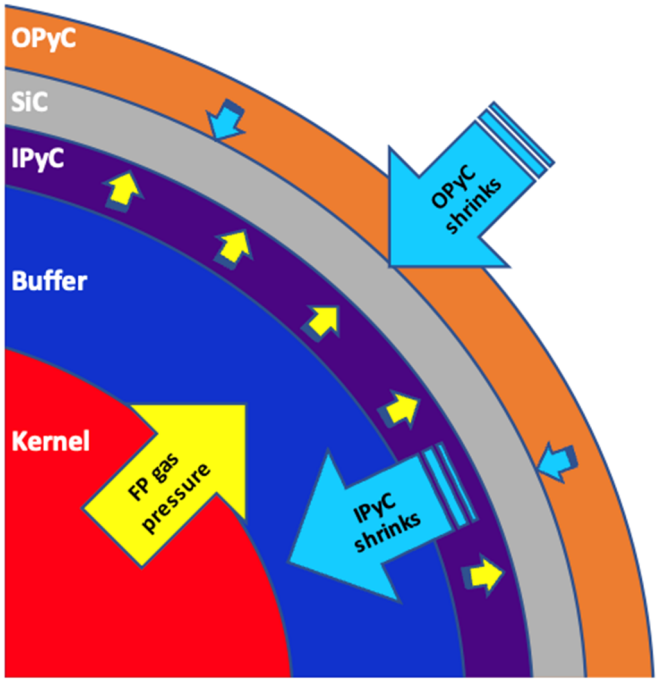
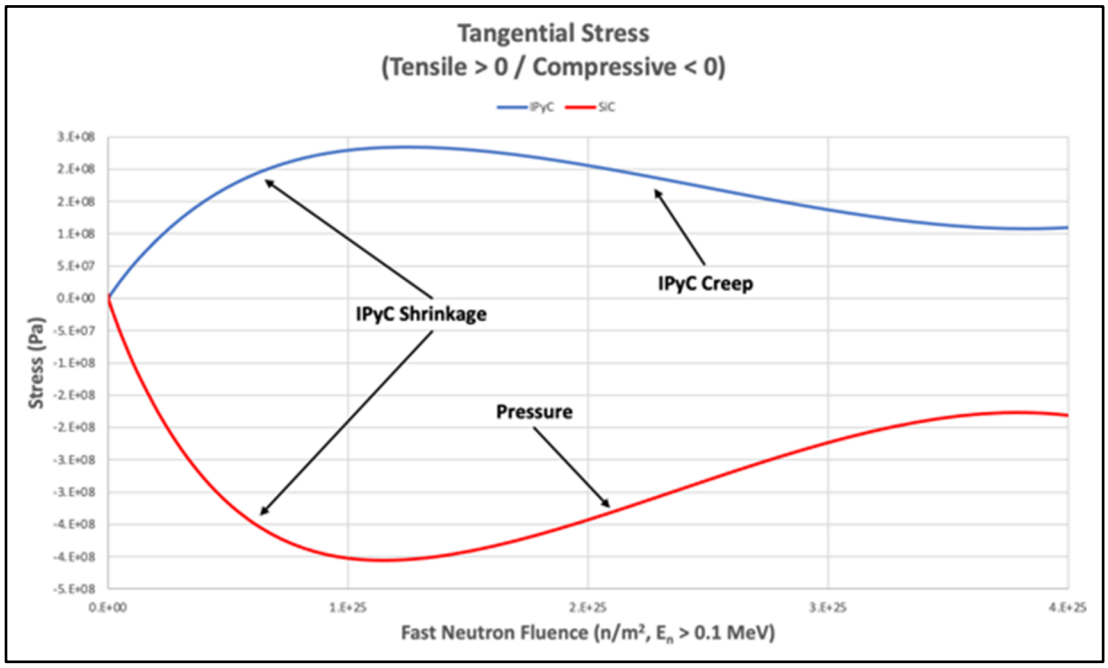


Table 3.1. *Irradiation Behavior of TRISO Particle Components*

Component	Irradiation Behavior
Kernel	Swells outward
Buffer	Shrinks inward
PyC	Shrinks at low fast neutron fluence Swells at moderate (radial) and high (tangential) fast neutron fluence levels
SiC	Elastic behavior

Figure 3.2. *Illustrative Stress in IPyC and SiC Layers of an Intact UCO TRISO Particle*



Fission product release to the helium coolant may occur from:

- Uranium contamination (referred to as “tramp” or “dispersed” uranium) outside of the SiC layer
- Diffusion of a few specific radioisotopes through intact TRISO particle layers
- Fabrication defects, or
- In-service failure

During fuel manufacturing, uranium removed by chloride attack of the fuel kernel during coating deposition or uranium from contaminated process equipment may be deposited on or outside of the SiC layer. Because the SiC provides the primary TRISO barrier to fission product release, fission products formed outside of the SiC layer are more easily released from the TRISO particle. Because of the very high integrity of TRISO particles, the fission products from the dispersed uranium are expected to be a substantial portion of the radioactivity released from the fuel early in the life of TRISO fuel. {{

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3.2 Fission Product Transport through Intact TRISO Layers

The various radioisotopes created by fission are in two atomic mass groups: 90-100 and 130-140 and include over 30 elements. The specific radioisotopes have different nuclear (amount produced, half-life, energy of emitted radiation, neutron cross-section), physical (gas or solid), and chemical (reactivity with other materials) properties that determine their radiological importance. Most radioisotopes produced by fission have low yields, short half-lives, or form low-mobility oxide compounds. Radionuclides that are not released from the UCO kernel are usually

excluded from fission product transport analysis. Conversely, fission products of radiological importance that are released from the kernel and can diffuse through the coating layers of the TRISO particles and through the fuel pellet are divided into relatively short-lived gaseous and long-lived metallic fission products in source term analysis.

3.2.1. Radioisotopes of Interest

Historically, the difficulty of evaluating the production, transport, and release of all species produced by fission in a TRISO kernel led to establishing a reduced list of radioisotopes of radiological importance for consideration in HTGR source term analysis [47]. For HTGRs, the key radioisotopes were selected through a combination of sufficient fission yields, ability to transport and release, and importance of radiological hazard. They are long-lived silver (^{110m}Ag), cesium (^{134}Cs, ^{137}Cs), strontium (^{90}Sr), and krypton (^{85}Kr) isotopes and the short-lived iodine (^{131}I) and xenon (^{133}Xe) isotopes. Isotopes of the same chemical species are assumed to have the same transport and release properties.

Some release characteristics of these radioisotopes under HTGR conditions are given below:

- Silver is released from intact TRISO particles at temperatures above 1000 °C; it plates out on metallic surfaces at temperatures lower than ~800 °C and on graphite at temperatures lower than ~900 °C; silver represents an occupational hazard during maintenance, but it is not considered an off-site radiological hazard.
- Cesium is released from TRISO particles with defective or failed SiC layers.
- Strontium is essentially retained in UCO kernels at normal HTGR operating temperatures.
- Iodine, krypton, and xenon have similar transport and release properties, and are traditionally grouped together; they are well retained by the PyC and SiC layers at HTGR normal operating temperatures.

3.2.2. Fission Product Transport Phenomena

The transport of mobile fission products through a TRISO particle is a complex process that depends on the microstructure of the kernel and coating layers. It can involve several mechanisms such as lattice diffusion, grain boundary diffusion, pore diffusion, nano-cracking, and vapor transport [48]. These potential transport mechanisms can also be impacted by effects such as irradiation-induced trapping and adsorption, thermal decomposition of the coating layers, or chemical attack of the coating layers by noble metals or rare-earth elements.

The transport of gases and metals in the kernel and coating layers is likely driven by different basic mechanisms. The fundamental knowledge of all transport phenomena is limited, however, and fission product transport in TRISO fuel is modeled by Fickian diffusion using effective diffusion coefficients. The effective diffusivities were derived by fitting experimental data that involve these transport phenomena, which makes them adequate to model transport. The phenomenon of diffusion is dependent on time at temperature – {{

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The influence of the SiC microstructure on the effective diffusivities in the SiC layer was evaluated in reference [49]. Electron backscatter diffraction was used to show that layer formation by chemical vapor deposition (CVD) at higher temperature (1500 °C) and using pure hydrogen and MTS results in a gradient in grain size and shape, with finer and more equiaxed grains present at the PyC/SiC interfaces and larger, more columnar grains in the center of the SiC layer. The finer grains at the IPyC/SiC interface are attributed to initial nucleation and growth of the SiC layer, while the reduction in grain size, relative to the mid-layer, at the SiC/OPyC interface is attributed to the termination of the growth of the SiC layer. Coating at lower temperature (1425 °C) using a mixture of argon and hydrogen with MTS produced a more equiaxed and finer grain structure in the SiC layer.

Fission product transport in polycrystalline solids such as the TRISO SiC layer can be modeled using an effective diffusivity that assumes that grain boundary diffusion is dominant. The effective diffusivity thus depends on the grain boundary area available for transport and the orientation of the grain boundaries relative to the direction of transport. The work in reference [49] compared measured fission product release from coarse grain, high-aspect ratio SiC microstructures to fine grain equiaxed microstructures. During irradiation and safety testing at 1600 °C and 1700 °C, no differences in fission product release behavior were noted between the two types of microstructures. At 1800 °C, higher release rates for ^{110m}Ag and ^{154}Eu were measured from TRISO particles with a fine-grained equiaxed microstructure, consistent with trends calculated using the Maxwell Garnett effective medium approximation.

3.3. TRISO Coated Particle Failure Mechanisms

Based on historical irradiation experience, the following failure mechanisms of TRISO-coated fuel particles have been identified [50] [1]:

- Pressure vessel failure of spherical or aspherical particles resulting in the failure of all three coating layers
- Irradiation-induced cracking of the IPyC layer leading to SiC failure
- Irradiation-induced partial debonding of the IPyC from the SiC leading to SiC failure
- Irradiation-induced cracking of the OPyC layer leading to SiC failure
- Kernel migration towards the SiC layer and its subsequent failure
- Chemical attack of the SiC layer by fission products, CO, or heavy metal contamination leading to its failure
- Thermal decomposition of the SiC layer at high temperatures
- Irradiation-induced buffer fracture leading to cracking of partially or fully attached IPyC

Additionally, potential OPyC cracking due to interaction with the surrounding FCM pellet matrix is discussed in Section 3.3.4.

3.3.1. Pressure Vessel Failure

Because of design and geometry, the SiC layer of the TRISO particle constitutes a thin-walled spherical pressure vessel. Internal pressure inside the SiC pressure vessel increases as fission gases produced in the kernel accumulate in the void volume provided by the porous buffer and the buffer-IPyC gap. Failure of the SiC layer occurs when tangential tensile stress of the SiC layer exceeds its fracture strength. Failure by internal pressure is more likely to occur late in irradiation and/or at high burnup when sufficient fission gas has been released into the void volume of the TRISO particle to build up internal pressure.

Production of Carbon Monoxide

Carbon monoxide (CO) is produced by the reaction of a net excess of oxygen with the carbon in the coating layers. As UO_2 undergoes fission, part of the released oxygen reacts with fission products to form oxide compounds, but these fission products are not thermochemically capable of binding all the liberated oxygen [50]. Oxygen not consumed by fission products can then oxidize the carbon internal to the SiC layer to form CO gas.

The amount of CO produced is a function of temperature and burnup and it depends on the composition of the kernel. CO production can be significant in UO_2-type fuels. UCO kernels are designed to limit CO production by tailoring the amount of carbon present in the kernel as UC and UC_2 phases. As UO_2 fission liberates oxygen, some of this oxygen can react with the UC and UC_2 compounds to convert them into UO_2, thus limiting the amount of free oxygen available to form CO by reaction with the buffer.

PIE of AGR-1 and AGR-2 test specimens did not find evidence of CO corrosion or failure driven by CO corrosion in UCO fuel after irradiation or safety testing. In contrast, AGR-2 UO_2 fuel showed substantially higher failure rates and direct evidence of CO corrosion during safety testing [43]. This data indicates that a negligible amount of CO is produced in UCO during the testing of TRISO particles with carbon-to-oxygen ratios in the range of the AGR-1 and AGR-2 irradiation tests.

SiC Tensile Stress Metric

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A so-called “SiC tensile stress metric” is described in reference [1] to compare the stress level in the SiC layer as a function of kernel diameter, buffer thickness, and maximum burnup. This metric evaluates the internal pressure in the TRISO particle and translates it into the stress in the SiC layer considered as a thin-wall pressure vessel.

Although this metric provides a simple tool to assess TRISO fuel performance, it assumes that the stress in the SiC layer is generated by internal pressure only and does not include the effects of PyC irradiation-induced dimensional changes and creep. In particular, PyC shrinkage at lower irradiation temperatures induces a higher compressive stress on the SiC layer. Furthermore, lower irradiation temperatures result in lower internal pressure for an equivalent void volume and fission gas inventory between fuel designs. Finally, the metric does not discriminate between UCO and UO_2 fuel types, and does not take into consideration that, everything else being equal, internal pressure is lower in UCO fuel compared to UO_2 fuel because of the absence of CO contributing to the internal pressure. As a result, the SiC tensile stress metric is not an adequate tool to assess the impact of the TRISO geometry on its SiC stress level. For FCM TRISO fuel, the comparison of stress level in the SiC layers to the AGR-1 and AGR-2 fuel designs would have to be performed with a TRISO fuel performance code that models all the phenomena contributing to stress in the SiC layer.

3.3.2. IPyC Cracking

IPyC cracking occurs in a TRISO particle when irradiation-induced shrinkage of the IPyC layer induces a tangential tensile stress that exceeds the fracture strength in that layer. Consequently, a radial crack can develop in the IPyC layer and propagate to the SiC interface. The subsequent loss of compressive stress from the IPyC layer onto the SiC layer results in tensile stress at the tip of the crack that can then lead to failure of the SiC layer if this tensile stress exceeds the SiC fracture strength.

IPyC cracking is a complex phenomenon that is dependent on irradiation temperature, fast neutron fluence, PyC density, and PyC anisotropy. In particular, more isotropic PyC is less prone to irradiation-induced dimensional changes and less prone to cracking.

IPyC cracking does not necessarily lead to mechanical failure of the SiC layer, because the fracture strength of SiC is usually higher than the tensile stress introduced by cracking of the IPyC. Additionally, debonding between the IPyC and SiC layers (Section 3.3.3) prevents any deleterious effect of IPyC cracking onto the SiC layer, but it constitutes a failure mechanism in itself. IPyC cracks also provide pathways for fission products to reach the SiC layer. The subsequent chemical attack on the SiC can then lead to its failure.

Cracking of the IPyC layer, and potential subsequent SiC failure, tends to occur early during irradiation when shrinkage stresses are maximum.

3.3.3. IPyC-SiC Debonding

IPyC-SiC debonding refers to the detachment of the IPyC and SiC layers due to tensile stress generated at their interface by irradiation-induced IPyC shrinkage. Debonding, or partial debonding, occurs when the radial tensile stress at the IPyC/SiC interface exceeds the bond strength between the two layers.

Debonding occurs as a progressive (partial) unzipping of the IPyC and SiC layers. It is usually initiated at a weak point of the IPyC/SiC interface and then progresses during irradiation. The tensile stress concentration created along the debonded path parallel to the IPyC/SiC interface is typically not as severe as the tensile stress created at the tip of a radial through-layer crack in the IPyC layer, but it affects a larger portion of the SiC surface [51].

Debonding between the IPyC and SiC layers, and potential subsequent SiC failure, tends to occur early during irradiation when the stress caused by IPyC shrinkage is maximum.

3.3.4. OPyC Cracking

The purpose of the OPyC layer is to keep the SiC layer in compression and protect it from external chemical attack.

The OPyC layer exhibits a similar behavior as the IPyC layer during irradiation, but at relatively lower stress levels. Consequently, it is less prone to failure and more able to keep the SiC layer under compressive stress. No evidence of OPyC cracking was reported from AGR-1 PIE. AGR-2 PIE identified OPyC cracks postulated to have formed during fuel fabrication [52] [43]. In the case of MMR FCM fuel, the interaction between the pellet matrix and the OPyC layer during fuel fabrication and irradiation is not expected to induce any OPyC cracking.

Gaps between the SiC and OPyC layers were observed in AGR-1 and AGR-2 PIE. The gap seems to form due to irradiation-induced shrinkage of the surrounding pellet matrix. As it shrinks outwards, the matrix pulls the OPyC layer away from the SiC layer. This results from the stronger bond between the matrix and OPyC due to interlocking of the two more porous interfaces, while the denser SiC forms a relatively weaker interfacial bond.

The presence of the SiC-OPyC gap means that the OPyC layer may not always keep the SiC layer under compression, but this does not seem to adversely impact fuel performance [43]. The presence of the gap may prevent OPyC cracks from propagating into the SiC layer by being disconnected at the debonded SiC/OPyC interface. {{

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3.3.5. Kernel Migration

Kernel migration, also called the *amoeba effect*, is the displacement of the kernel inside the TRISO particle under the influence of a macroscopic temperature gradient. In effect, the kernel is pushed towards the hot side of the TRISO particle by carbon dioxide (CO_2) and solid p-hase carbon (C) produced on the cold side of the particle by CO migrating down the temperature gradient and reacting as $2CO \rightarrow CO_2+C$. Particle failure is assumed to occur when the kernel comes into contact with the SiC layer. Kernel migration is more prominent in UO_2 kernels than in UCO kernels because of the higher level of CO produced by the reaction between UO_2 and carbon in the TRISO particles. Kernel migration was not observed during PIE of AGR-1 and AGR-2 UCO particles [52] [43].

3.3.6. Chemical Attack of the SiC Layer

Degradation of the SiC layer can result from the chemical attack by noble metal fission products, lanthanide fission products, and carbon monoxide.

Chemical Attack of the SiC Layer by Noble Metal Fission Products

Noble metals produced by fission can be transported from the kernel to the inner surface of the SiC, as thermochemical conditions during irradiation do not permit the formation of stable oxides in the kernel [50] [1].

For instance, palladium has been observed to migrate to the SiC layer where it can threaten its integrity by reacting with SiC to form palladium silicides. As such, Pd is regarded as a major contributor to the attack of SiC and to penetration into the SiC layer, resulting in potential loss of fission product retention capability. AGR-1 PIE has observed Pd release from TRISO particles through intact SiC, but no corrosion or attack of SiC was observed on these as-irradiated particles [1]. Pd corrosion of the SiC layer is postulated to only occur when IPyC cracks provide pathways for Pd to locally concentrate at the IPyC/SiC interface [52]. However, AGR-2 PIE identified shallow isolated regions of SiC degradation with no signs of cracked or degraded IPyC [43]. There was also evidence of SiC degradation in AGR-2 TRISO particles initiated at the SiC/OPyC interfaces, which was attributed to external attack by nickel from nearby thermocouples. Should Pd be able to locally concentrate at the IPyC/SiC interface, complete penetration of the SiC layer would require that high temperatures are maintained for a sufficient time (e.g., 1600 °C for 400 days, 1300 °C for 1250 days, or 1000 °C for 7000 days). {{

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Chemical Attack of the SiC Layer by Lanthanide Fission Products

Lanthanide and rare-earth fission products are strongly retained in UO_2 where they form stable and low-mobility oxide compounds [53]. Proper balancing of the oxygen and carbon proportions during manufacturing of UCO kernels allows full oxidation of lanthanide and rare-earth fission products during irradiation and mitigates their potential attack and corrosion of the SiC layer. The specified C/U and O/U ratios for FCM TRISO fuel are similar to the ratios in the AGR-1 and AGR-2 UCO fuel. At these ratios, the UC_2 content of UCO is ~20%, which favors the formation of lanthanide oxide compounds over lanthanide carbides to burnups of ~18 %FIMA. Lanthanide oxides have lower mobility and increased retention in the kernel [53].

Chemical Attack of the SiC Layer by Carbon Monoxide

Excess oxygen produced in the kernel can react with carbon in the buffer to produce CO. This chemical reaction is more prominent at higher burnup as the swelling kernel becomes more porous to oxygen release. CO produced within the TRISO particle contributes to the internal pressure but also to potential corrosion of the SiC layer. Chemical attack of the SiC layer, hypothesized to be from CO corrosion, was observed by AGR-2 PIE in UO_2 kernels safety-tested at temperatures above 1700 °C, but it was not identified for either type of kernel at irradiation temperatures or for UCO at safety testing temperatures [1]. The nominal oxygen content of UCO in FCM TRISO fuel will be similar to the initial content in the AGR-1 and AGR-2 UCO fuel, which will limit CO production. {{

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Thermal Decomposition of the SiC Layer

During exposure for long times at very high temperatures, SiC thermally decomposes back into its constituent elements. Silicon vapor then migrates out of the SiC coating, which leaves behind a porous carbon layer that is not retentive of fission products.

Initial signs of SiC decomposition and SiC porosity have been observed at temperatures near 1800 °C although decomposition mainly occurs at temperatures above 2100 °C [54]. PIE of the AGR-1 and AGR-2 irradiation tests showed no signs of SiC decomposition after irradiation at time-average peak temperatures of 1197 °C and 1360 °C, respectively.

Safety testing of AGR-1 and AGR-2 fuel at temperatures of 1800 °C also did not provide any evidence of SiC decomposition. {{

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3.3.7. Cracking of Partially Debonded Buffer/IPyC Layers

PIE of AGR-1 and AGR-2 TRISO particles identified a mechanism suspected to be responsible for observed SiC failures in TRISO particles in which the buffer and IPyC layers did not fully

delaminate during irradiation. In this scenario, the shrinking buffer layer transfers stress to the IPyC layer through the remaining bonded region, causing IPyC fracture. Subsequent transport of Pd through the IPyC crack results in localized accumulation at the IPyC/SiC interface. Pd accumulation at the IPyC/SiC interface then results in degradation of the SiC layer [52] [43].

As noted in [52], AGR-1 TRISO particles exhibiting failed SiC showed evidence that cracked IPyC had exposed the inner surface of the SiC, which allowed accumulation of fission products that chemically degraded the SiC structure. Widespread chemical attack of the SiC layer by Pd was not observed in the absence of IPyC cracking. Additionally, SiC failure usually followed a similar mechanism at safety-testing temperatures, but at an accelerated rate.

Ceramography of irradiated AGR-2 fuel did not identify clear evidence of through-layer IPyC fractures associated with partial buffer-IPyC debonding, but some SiC degradation was observed. It is likely due to accumulation of fission products at the IPyC-SiC interface through an IPyC crack that developed along the edge of buffer detachment [43].

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3.4. FCM Fuel Pellets

The purposes of the FCM fuel pellet are to:

- Provide an additional barrier to release of fission products to the coolant
- Isolate TRISO particles from interaction with the graphite fuel element block
- Transfer heat from the TRISO particles to the pellet-block interface

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}} The FCM pellet structure serves to isolate the TRISO particles from interaction with the graphite fuel block and from chemical interaction with coolant impurities.
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3.4.1. Properties of FCM Pellets

SiC material used for nuclear applications is produced by the reduction of chemical precursors during high-temperature CVD and CVI processes. CVD processes generally refer to SiC deposition on a surface, while CVI processes refer to deposition within the structure of a component. CVD and CVI processes used for deposition of SiC use the same chemical precursors (MTS and hydrogen), but may differ in temperature, atmosphere (pressure or vacuum), and concentration

of precursors. SiC produced by both CVD and CVI methods have high purity, nearly exact SiC stoichiometry, and are crystalline. These attributes are critical for attaining predictable swelling behavior [55] and for preventing degradation in strength [56] from the effects of irradiation displacement damage. CVD SiC mechanical and thermal properties and their dependence on irradiation are reported in reference [57].

{{
}} The differences in processing conditions may affect the grain size of the deposited SiC, but do not affect its purity, stoichiometry, or crystallinity.

Thermal and mechanical properties of the FCM CVI SiC matrix (e.g., characteristic strength, Weibull modulus, thermal conductivity) were measured on unirradiated material and material irradiated to 2.3 displacements per atom (dpa) [40]. Mechanical properties showed little variation in strength or changes in microstructure at different temperatures or printing orientation. The thermal conductivity of FCM SiC exhibited a strong dependence on irradiation temperature but no dependence on orientation for irradiated material. Measurements of FCM-SiC properties specific to the FCM fuel pellet will be made as part of the Fuel Qualification Program to provide additional data on thermal and mechanical properties.

Under neutron flux, the swelling behavior of CVD SiC varies across three temperature regimes [57] [58]:

- At irradiation temperatures lower than approximately 150 °C, the accumulated strain caused by irradiation defects can lead to amorphization and high swelling rates.
- At temperatures between 150 and approximately 1000 °C, CVD SiC swelling in the point defect regime occurs at a rate that increases logarithmically with fast neutron dose until it reaches a saturation level. The saturation level at which swelling stops decreases with increasing irradiation temperature.
- At temperatures above the point defect swelling regime, CVD SiC exhibits dose dependent void swelling behavior in which vacancy clusters can coalesce into three-dimensional cavities or voids.

Based on current experimental data, the irradiation behavior of FCM CVI SiC is expected to be similar to other high-purity and polycrystalline SiC, such as CVD substrates commercially used in the electronics industry that form a significant portion of the irradiation behavior database. In particular, CVI SiC is expected to have similar irradiation-induced swelling behavior. The swelling of CVD SiC increases logarithmically with fast neutron dose until saturation. Saturation occurs at relatively low neutron dose, at which point swelling stops. CVD SiC swelling is also dependent on temperature, exhibiting higher swelling at saturation during irradiation at lower temperatures.

{{
}} Although unirradiated CVI SiC has relatively high thermal conductivity, it degrades with fast neutron fluence [40]. As reported in reference [59], in the point defect regime, the thermal conductivity of irradiated CVD SiC decreases and saturates at relatively low dose (a few

dpa's). Because scattering from point defects dominates thermal transport under these conditions, the dependence of thermal conductivity on temperature is reduced. The specific heat capacity of CVD SiC is reported to be negligibly affected by neutron irradiation [60].

3.4.2. FCM Fuel Performance

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SiC acts as the primary barrier to fission product release in TRISO fuel particles. The MMR FCM fuel uses SiC in two different forms to (1) provide a fission product barrier in each TRISO particle and (2) provide an additional barrier on each fuel pellet that retains fission products released by TRISO particles. This approach provides defense-in-depth to functional containment. The fission product release behavior of SiC has been characterized by multiple international programs [61]. Diffusion coefficients for groups of fission products are represented as a function of temperature in mathematical functions.

A comparison of the accuracy of the PARFUME code [38] to experimentally measured fission product release was performed using data from the AGR-1 irradiation test [62]. The AGR-1 test was irradiated to a peak burnup of 19.6 %FIMA and a fast neutron fluence of 4.3×10^{25} n/m^2 (E > 0.18 MeV) over 620 EFPD [25]. Fission product release values were calculated based on measured daily fuel temperatures that ranged from < 500 °C to 1500 °C over the test duration. The comparison showed that PARFUME overpredicted cesium and strontium release, indicating that diffusion coefficients used in PARFUME are conservative for the conditions tested. Of less consequence to off-site dose was the release of silver, which was underpredicted at low temperature and overpredicted at high temperature.

{{

}} Cs diffusion in SiC is modeled as Fickian diffusion using a diffusivity that follows an Arrhenius law:

$$D = D_1 \times e^{-\frac{Q_1}{R\times T}} + D_2 \times e^{-\frac{Q_2}{R\times T}}$$

Where "D" (m^2/s) is the diffusivity, "D_1" (m^2/s) and "D_2" (m^2/s) are pre-exponential diffusion coefficients, "Q_1" and "Q_2" are activation energies, "R" (8.3145 J/mol-K) is the ideal gas constant, and "T" (K) is the {{

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Figure 3.3. {{ }}

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Table 3.2. *Coefficients for Cs Diffusivity SiC used in PARFUME*

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3.4.3. Potential FCM Failure Modes

Because SiC is a brittle material, failure strength is a consequence of the distribution of flaws present in the sample relative to the distribution of stress. Fracture will occur at a location where the tensile component of the stress is large enough to propagate a crack from a flaw of critical size. This type of behavior is described by Weibull methods, in which crack-initiating flaws are assumed to be randomly distributed throughout the component surface or volume. The probability of failure "P_f" for the two-parameter Weibull distribution is written as:

$$\mathrm{P_f} = 1 - e^{-(\frac{\sigma_f}{\sigma_0})^m}$$

Where "σ_f " is the applied stress, "σ_0" is the characteristic strength, and "m" is the Weibull modulus. Values of "m" and "σ_0 " are determined from experimental data. {{

}}

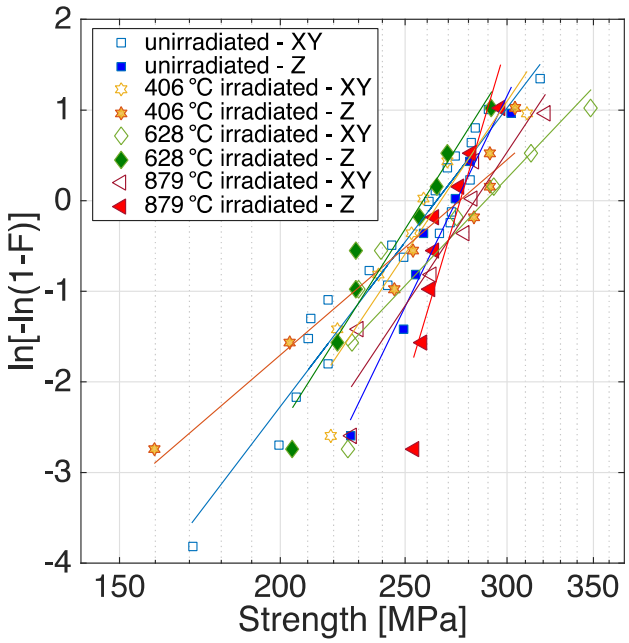
Preliminary measurements of the Weibull parameters of irradiated and unirradiated SiC specimens fabricated using the same process as FCM fuel pellets [40]. These data are shown in **Table 3.3** and **Figure 3.4**. {{

}}

Table 3.3. *Characteristic Strength and Weibull Modulus of FCM SiC Specimens from* ***Figure 3.4***

Irradiation	Orientation	Characteristic strength (MPa)	Weibull modulus (m)	Number of tests
Unirradiated	XY	264	8	23
	Z	274	13	7
Irradiated (SDTR01)	XY	266	9	7
	Z	276	5	8
Irradiated (SDTR02)	XY	289	7	8
	Z	258	10	8
Irradiated (SDTR03)	XY	283	9	7
	Z	276	21	8

Figure 3.4. *Weibull Plot of FCM SiC Strength for Irradiation to 2.3 dpa*



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SiC is among the hardest of ceramic materials, with a measured Vickers hardness ranging from 20.7 – 24.5 GPa [57]. {{

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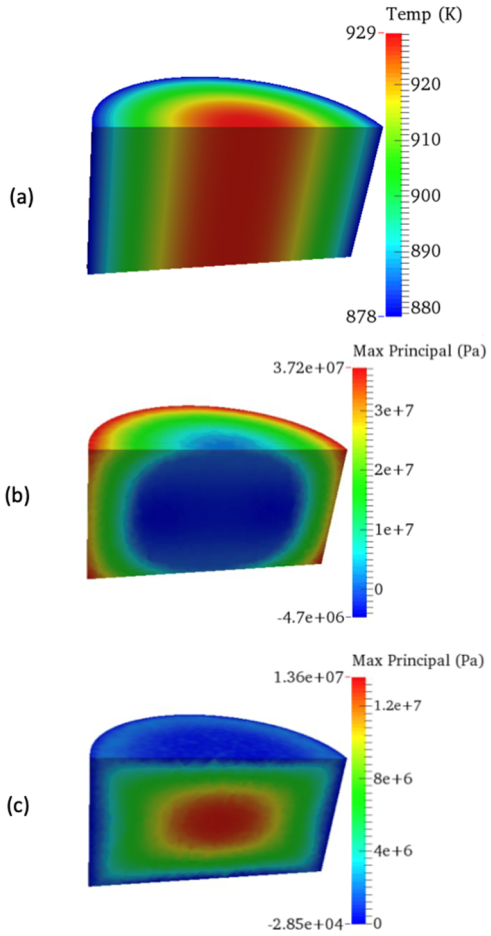
An illustration of this behavior is shown in **Figure 3.5**, from reference [63]. The example assumes an FCM fuel pellet with the constant pellet temperature distribution shown in **Figure 3.5** (a). During initial reactor start up, in the absence of irradiation-induced swelling, pellet stress is driven by the thermal gradient resulting from heat generation from fissions within the embedded TRISO particles. The larger thermal expansion in the higher temperature central region of the pellet matrix generates a tensile stress on the pellet outer surface. **Figure 3.5** (b) indicates the maximum principal stress at the beginning-of-irradiation is tensile and near the pellet surface. {{

}} **Figure 3.5** (c) shows the maximum principal stress after saturation of swelling. The stress on the outer pellet surface becomes compressive while the stress in the central region of the matrix becomes tensile.

Figure 3.5. *Illustration of the Evolution of the Stress in FCM Pellets as a Function of Accumulated Fast Neutron Fluence*



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The helium coolant of the MMR core will be contaminated by small amounts of gaseous impurities (e.g., H_2, H_2O, CH_4, CO, CO_2, O_2) from the original gas supply and from a variety of sources in the primary circuit. Impurity levels are expected at the parts per million level, similar to previously operated HTGRs [64]. Because of the large surface area of graphite in HTGR cores, gaseous oxygen exists at very low levels in HTGR core.

Oxidation of the FCM fuel pellets and TRISO particles could occur if exposed to these impurities at high temperatures. {{

}} The MMR utilizes two systems to establish and maintain low impurity levels in the helium coolant:

- The function of the Primary Coolant Clean-up system is to satisfy the criteria for purity of the helium in the vessel prior to MMR start-up. Helium purity is achieved by several cycles of evacuation of the vessel system and back filling with pure helium.

- The function of the Helium Purification System is to remove dust and chemical impurities to maintain the required coolant chemistry. The helium is cleaned in a side stream flow

taken from the intermediate heat exchanger vessel and returned in a separate line to the intermediate heat exchanger vessel.

Oxidation of SiC may occur in passive and active regimes. Passive oxidation of SiC results in the formation of a protective silicon oxide surface layer that slows diffusion of oxygen to the surface.

$$\mathrm{SiC} + \frac{3}{2}\mathrm{O_2} \rightarrow \mathrm{SiO_2} + \mathrm{CO}$$

Passive oxidation is characterized by mass gain. During active oxidation, ambient oxygen reacts with SiC to form volatile SiO and CO.

$$\mathrm{SiC} + \mathrm{O_2} \rightarrow \mathrm{SiO} + \mathrm{CO}$$

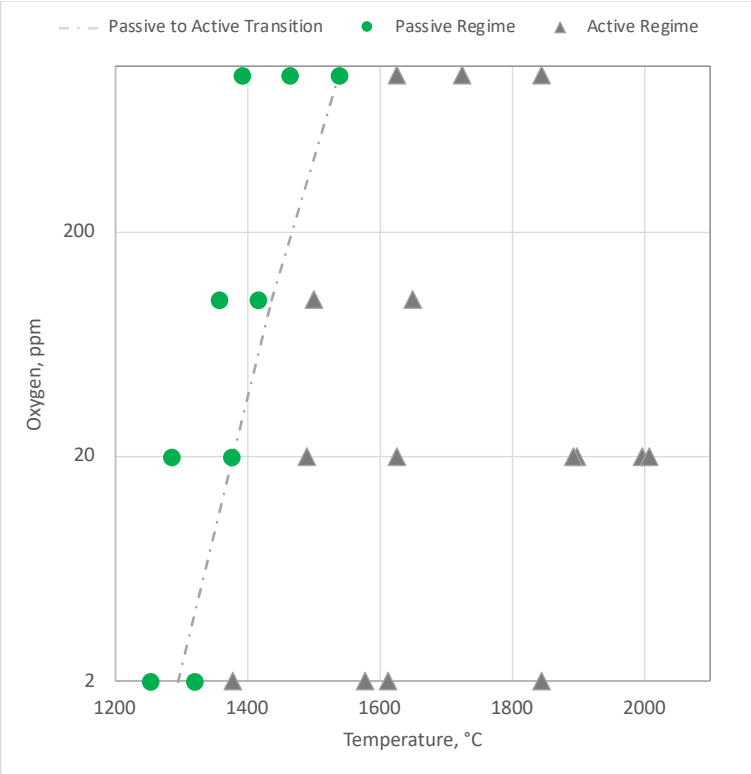
Active oxidation occurs if ambient conditions do not allow the formation of a protective silica surface layer, for example at very low oxygen levels and high temperatures. The rate of SiC mass loss in active oxidation regime decreases as the partial pressure of oxygen decreases.

Oxidation testing of CVD SiC was carried out at 900 °C and 950 °C in a simulated HTGR coolant environment (measured partial pressures of 20 Pa H_2, 5 Pa CO, 2 Pa CH_4, 0.02 - 0.1 Pa H_2O) with negligible measured O_2 content [65], similar to the peak fuel temperatures and coolant impurity levels expected during MMR operation. The surface of the SiC was examined using X-ray Photoelectron Spectroscopy after exposures of 25, 50, 100, and 250 hours, indicating the growth of a thin (< 10 nm) silica layer. The observed passive oxidation behavior was attributed to the presence of low levels of impurity H_2O.

Oxidation testing of CVD SiC and TRISO particles at temperatures representative of MMR accident conditions has also been conducted in oxygen containing inert gas and steam environments. Steam oxidation tests designed to determine behavior under LWR accident conditions initially suffered from contamination of the protective silica scale by impurities transferred from aluminum oxide reaction tubes. Replacement of aluminum oxide furnace tubes with zirconium oxide or SiC tubes eliminated this issue.

The high temperature oxidation behavior of CVD SiC was experimentally determined as a function of temperature (1257 – 2009 °C) and oxygen impurity levels (2 – 2000 ppm) to determine the transition from passive to active oxidation [66]. **Figure 3.6** plots the experimentally observed transition from passive to active oxidation, which occurs near 1300 °C at low oxygen levels (2 ppm). Measured mass loss during active oxidation was near zero below 1450 °C for the entire range of oxygen concentrations from 2 – 1000 ppm.

Figure 3.6. *Experimental Data from High-temperature Air Oxidation of CVD SiC*



Oxidation tests of FCM SiC were conducted in both flowing steam and air environments at 1300 °C and 1425 °C for isothermal hold times of 100 hours in SiC reaction tubes [67]. Thin crystalline silica (SiO_2) layers form during both air and steam oxidation, with mass gains and oxide layer thickness shown (**Table 3.4**), indicating passive oxidation in both cases with moderate oxide layer thickness.

Table 3.4. *Experimental Data from Steam and Air Oxidation Testing of FCM SiC*

Oxidation Temperature (°C)	Oxidation Environment	Mass Change After 100 hours (mg/cm^2)	Oxide Layer Thickness (μm)
1300	Air	0.09	1.1
1300	Steam	0.53	7.1
1425	Air	0.22	3.4
1425	Steam	0.82	9.4

The behavior of the CVD TRISO SiC layer evaluated using flowing steam and air at 1 atmosphere pressure and temperatures of 1500 – 1700 °C [68]. Test durations were varied at 4 and 24 hours to determine the time dependance of oxidation. Oxide layer thickness and SiC layer thickness were measured after testing, the latter to determine recession of the SiC layer. Data is provided in Table 3-5. After testing, SiC layer thickness showed small changes within a few standard deviations of the as-fabricated particles. Oxide layer thickness was larger during steam oxidation, with a maximum oxide layer thickness at 1600 °C of 6.9 μm relative to 3.1 μm for air oxidation.

Table 3.5. *Experimental Data from Steam and Air Oxidation of TRISO SiC*

Test Temperature (°C)	Test environment	Test duration (h)	Mean SiC layer thickness (μm)	SiC layer standard deviation (μm)	Oxide layer thickness (μm)
As fabricated	-	-	32.2	0.7	0
1500	steam	4	30.2	0.9	2.3
1550	steam	24	32.4	1.1	4.4
1600	steam	4	30.1	0.9	3.1
1600	steam	24	29.7	1.0	6.9
1600	air	24	33.6	1.5	3.1
1650	steam	24	31.9	0.9	3.7
1700	steam	4	30.8	1.0	7.2

The experimental data represented in **Figure 3.6**, **Table 3.4**, and **Table 3.5** indicate that damage to the FCM pellets and TRISO fuel by oxidation is unlikely for the range of temperatures analyzed for MMR normal operation, AOOs, and DBAs.

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4. FUEL MANUFACTURING AND QUALITY CONTROL

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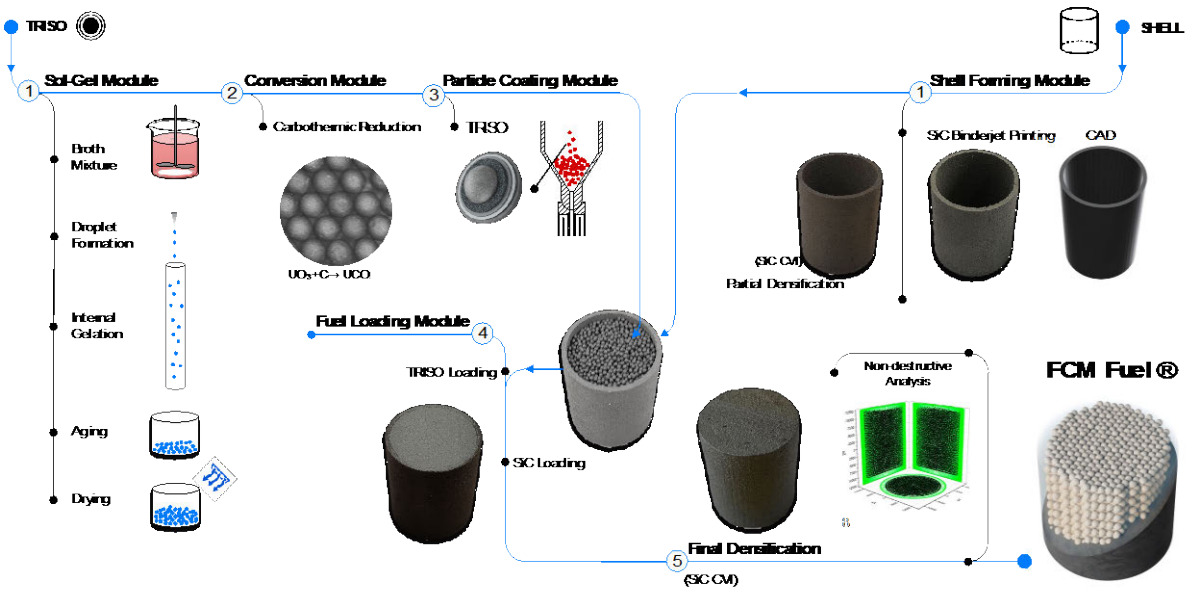
}} The fabrication process for encapsulation of TRISO particles in SiC to form FCM fuel pellets was developed by U.S. DOE's TCR program. **Figure 4.1** provides an overview of the FCM manufacturing process. The left side of **Figure 4.1** shows the process for TRISO particle manufacturing. The right, shaded side of **Figure 4.1** shows the FCM pellet manufacturing process, including incorporation of TRISO particles into the FCM pellet. The FCM fuel pellets used for fuel qualification will be manufactured using this equipment and processes in the USNC-owned and operated Pilot Manufacturing Facility located near Oak Ridge, Tennessee.

Figure 4.1. *Process Flow Diagram of FCM Fuel Manufacturing Process*



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}} AGR TRISO fuel was manufactured to a product specification that has been demonstrated to produce fuel particles with acceptable and repeatable fuel performance. {{

}}

TRISO particles produced by the AGR program using different fabrication processes and manufacturing process equipment exhibited excellent performance during irradiation and safety testing. Ranges of process variables provided in this section are typical of those used to manufacture AGR TRISO particles to manufacture TRISO particles within specification limits. {{

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TRISO fuel particles are manufactured using a sequence of batch processes. Each of the three major TRISO manufacturing processes (steps 1 – 3) on the left side of **Figure 4.1** are conducted within manufacturing modules specifically designed for each process step. {{

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The TRISO fuel manufacturing process as implemented by the AGR program and adopted by USNC consists of three primary steps:

- Manufacturing of UCO precursor gel spheres using the Sol-Gel module
- Conversion of gel spheres to UCO kernels using the Conversion module
- Coating of UCO kernels with buffer, dense PyC, and SiC layers using the Particle Coating module

Each step in the TRISO manufacturing process includes sampling and quality control measurements (Section 4.2) to ensure that the manufactured TRISO fuel particles meet all product specifications. {{

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4.1.1. Manufacturing of UCO Precursor Gel Spheres Using the Sol Gel process

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}} Fabrication of AGR-1 TRISO SiC used only hydrogen gas at a temperature of 1500 °C. Dilution of hydrogen with argon and deposition at 1425 °C was used by the AGR program to produce a TRISO SiC layer with a finer and more equiaxed grain structure for irradiation of AGR-1 variant 3 and AGR-2 [73] [74]. Fuel for irradiation in the AGR-5/6/7 test was fabricated using a deposition temperature of 1565 °C using a mixture of argon and hydrogen with MTS [75].

{{

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4.1.2. Manufacturing of FCM Fuel Pellets

The MMR FCM fuel form is a cylindrical pellet composed of TRISO particles encapsulated in a SiC matrix. {{

}}

A “green” shell is fabricated using binder jet printing, where an aqueous binder is deposited by a print head onto a bed of SiC powder in a series of layers [76]. After printing, the green shell is removed from the powder bed, excess powder is removed, and the part is heated to cure the binder and to strengthen the part to allow handling. {{

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After partial densification, the shells are filled with UCO TRISO and additional SiC powder. The mass of TRISO particles loaded into each shell is determined by the specified uranium content of the FCM pellet. {{

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Partially densified FCM shells filled with TRISO particles and SiC powder are loaded onto graphite fixtures and placed in a CVI furnace system. {{

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4.2. Quality Control of TRISO particles

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}} USNC quality control procedures ensure that critical properties of the TRISO fuel particles as identified in reference [1] are within the specified USNC tolerances.

4.2.1. Statistical Sampling Methods

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Because of the large number of particles and the destructive nature of many of the quality control inspections, the entire population of particles cannot be individually inspected. {{

}} Acceptance of a particle lot is based on comparison of the attribute or variable properties of the particle sample to specified requirements. {{

}}

4.2.2. Quality Control of TRISO particles

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}} Key properties of TRISO particles inspected and controlled by USNC procedures are summarized in **Table 4.1**.

Table 4.1. *Key Properties of TRISO Particles Subject to Quality Control Procedures*

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4.3. Quality Control of FCM Pellets

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5. METHODOLOGY FOR FUEL QUALIFICATION

FCM fuel design relies on the previously demonstrated TRISO particle architecture [1] [2] and the FCM pellet as barriers to fission product release. {{

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Fuel qualification focuses on demonstrating the in-service performance of the integrated FCM fuel system, consisting of TRISO particles embedded in a SiC FCM pellets. Key performance parameters are radionuclide retention and pellet integrity.

Activities included in USNC's fuel qualification program include:

- Development of fuel product specifications for TRISO particles and FCM pellets (Section 2)
- Demonstration of fuel manufacturing and quality control processes capable of consistently meeting specifications (Section 4)
- Testing and characterization of unirradiated fuel and materials
- Fuel pellet irradiation tests in material test reactors (MTRs)
- Post-irradiation safety testing of irradiated fuel pellets to measure performance in simulated accident conditions
- PIE of fuel pellets after irradiation testing and after post-irradiation heat-up tests to determine fuel performance and material properties of irradiated fuel pellets
- Development and validation of fuel performance models

5.1. Testing and Characterization of Unirradiated Fuel and Materials

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5.2. Fuel Pellet Irradiation Tests in Material Test Reactors

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5.2.1. Types of Irradiation Tests

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5.2.2. Irradiation Testing Parameters

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5.2.3. Irradiation Test Articles

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5.2.4. Steady-state Irradiation Testing

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Table 5.1. *Parameters Targeted for Steady-State Irradiation Testing*

Test Parameter	Target Value	Notes
Peak pellet temperature	{{ }}	{{ }}
Intermediate fuel burnup test (1)	{{ }}	{{ }}
Intermediate fuel burnup test (2)	{{ }}	{{ }}
Peak fuel burnup test	{{ }}	{{ }}
Pellet Power	{{ }}	{{ }}
Fast neutron fluence	{{ }}	{{ }}
Type 3 pellet temperature	{{ }}	{{ }}

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5.2.5. Short-duration Irradiation Tests

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5.3. Post-irradiation Safety Testing

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5.4. Post-irradiation Examination

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5.4.1. Visual Inspection and Dimensional Measurement

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5.4.2. Immersion Density

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5.4.3. Acid Leaching

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5.4.4. Determination of TRISO Particle Failure

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5.4.5. Measurement of Thermal and Physical Properties

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5.5. Development and Validation of Fuel Performance Models

Fuel performance modeling is used to predict fuel performance during MMR normal operating conditions and AOOs, DBAs, and BDBAs for fuel design and safety evaluation.

Performance analysis of the MMR FCM fuel will be achieved using licensed fuel performance modeling codes. TP3 is being developed by USNC for analysis of TRISO particle performance, using models and calculation methods similar to the PARFUME code that is used as the basis for fuel performance modeling for the AGR program [38]. TP3 comprises stress and diffusion modules which can be run independently or coupled.

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5.5.1. Fuel Performance Model Inputs

The input parameters to TRISO fuel performance modeling codes include fuel (TRISO particle and FCM fuel pellet) characteristics and irradiation conditions. These are parameters to the material properties and physical models used by the codes to determine the mechanical state of the fuel. This mechanical state is then used to assess the mechanical integrity of the fuel barriers and their leak-tightness to fission product release.

Material properties and physical models have been developed over the years to properly model TRISO fuel performance. For MMR FCM fuel, these material properties and physical models will be developed by USNC.

The typical input parameters for MMR FCM fuel modeling are given below:

- UCO kernel: ^{235}U enrichment (wt%), O/U and C/U ratios, density (kg/m^3), diameter (m), and average grain radius (m)
- Buffer: density (kg/m^3), theoretical density (kg/m^3), and thickness (m)
- PyC: density (kg/m^3), BAF (-), and thickness (m)
- SiC: density (kg/m^3), thickness (m), and aspect ratio (-)
- Matrix: density (kg/m^3), radius (m), and packing fraction (-)
- {{ }}
- As-fabrication defects: dispersed uranium fraction (-), exposed kernel fraction (-), defective SiC coating fraction (-), and defective IPyC/OPyC coating fractions (-)

The input parameters for the MMR core irradiation conditions that are required for TRISO fuel performance modeling codes can vary from code to code but they relate to the following:

temperature (K) of the pellet surface (boundary condition for thermal heat transfer calculations), fission rate density (fission/m^3-s) in the kernel (used for burnup, heat generation, and fission product inventory calculations), and fast neutron flux ($\times 10^{25}$ n/m^2-s, E > 0.18 MeV) on the coating layers (used for fast neutron fluence calculation and irradiation-induced behavior). Depending on the application, such as design or safety analysis, different sets of irradiation conditions can be used; specific interest can be put on best estimate average core values or on extremum values to provide bounding envelopes.

5.5.2. Fuel Performance Model Outputs

The main outputs of TRISO fuel performance modeling are the time-dependent fuel failure probabilities and fission product release fractions. Specifically, the calculated failure probabilities are:

- Probability of particle failure
  - Contribution due to palladium penetration
  - Contribution due to IPyC cracking
  - Contribution due to IPyC-SiC debonding
  - Contribution due to internal pressure (including SiC asphericity)
- Probability of IPyC cracking
- Probability of IPyC-SiC debonding

The following time-dependent outputs are used to compute the main outputs and/or are provided for data analysis:

- Stress distributions
  - Radial stresses at IPyC/SiC and SiC/OPyC interfaces
  - Tangential stresses at inner and outer surfaces of IPyC, SiC, and OPyC layers
- Displacements and radii of kernel, buffer, buffer-IPyC gap, and outer coating layers
- Temperature distributions in kernel, buffer, buffer-IPyC gap, outer coating layers and pebble
- Fission gas inventory
- Internal gas pressure
- Release rate to birth rate ratios (R/Bs) of short-lived fission gases
- Fission product (non-gas) inventory

5.5.3. TRISO Particle Performance Program

TP3 is being developed by USNC. It comprises two distinct modules, the stress and diffusion modules, which can be run independently or coupled.

TP3 Stress Module

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TP3 Diffusion Module

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5.5.4. Ansys

Ansys is a commercial software package developed and marketed to provide engineering solutions and multiphysics simulation software for product design, testing, operation, and analysis . Its suite includes a wide array of products, including acoustics, electronics, fluids, optics, photonics, structures, etc.

{{

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5.5.5. Verification and Validation and Uncertainty Quantification

The modeling tools and methodology used by USNC to perform licensing activities will be subject to verification and validation (V&V) and uncertainty quantification (UQ), respectively:

- Verification is the process of determining that a model implementation accurately represents the developer's conceptual description of the model and the solution to the model. It is the assessment of the accuracy of the solution to a computational model by comparison of predicted values to known solutions. Therefore, it is tasked with identifying, quantifying, and reducing errors in the computational model and its numerical solution.

- Validation is the process of determining the degree to which a model is an accurate representation of the real world from the perspective of the intended uses of the model. It is the assessment of the accuracy of a computational simulation by comparison of predicted values to corresponding experimentally measured data.

- Uncertainty quantification (UQ) is the forward propagation of uncertainty to predict the overall uncertainty in model outputs. UQ tries to determine the likelihood of certain outcomes when some required physical parameters of the system are not precisely known. For these input parameters, assumption can either be that their uncertainties are characterized by probability distributions or that they correspond to conservative upper bounds.

Additionally, sensitivity analysis (SA) can be used to quantify the impact of the input parameters of a computational model on its output variables, i.e., to determine how variability of the input causes variability in the output. In the case of USNC modeling tools, SA will be used to identify the key parameters that have the most impact on the calculated outputs of the codes. As a result of SA, UQ can be restricted to these key parameters.

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5.5.6. Fuel Performance Modeling in Support of Design

Performance analysis of the MMR FCM fuel will support various aspects of design to validate choices in fuel specification and manufacturing. {{

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- *Palladium Penetration*

Since palladium penetration is an identified failure mode for the TRISO SiC, it will be evaluated {{ }} under conditions of normal operation, AOOs, DBAs,

and BDBAs. The depth of penetration will be compared to the thickness {{
}} to assess their potential loss of fission product retention capability.

5.5.7. Fuel Performance Modeling for Safety Evaluation

Fuel performance modeling will be used to assess the performance of the MMR FCM fuel during normal operation, AOOs, DBAs, and BDBAs. The two figures of merit used to evaluate TRISO fuel performance are in-service failure probability of the TRISO particles or fuel pellet and fission product fractional release from the fuel element. Consequently, these two figures of merit will be calculated for a population of TRISO particles and fuel pellets under expected in-service conditions. Additionally, based on the list of AOOs, DBAs, and BDBAs, the potential increase in failure probability or fission product release will be assessed during these AOOs, DBAs, and BDBAs using inputs for fission power density, neutron flux, and pellet surface temperature provided by relevant neutronics and thermal-hydraulics codes.

Additionally, sensitivity calculations will be performed to identify key parameters to fuel performance so that parameters with negligible impact are not restrained to ranges that prove too restrictive.

Similarly, sensitivity studies will be conducted on key parameters to fission product release from the TRISO particles.

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Upper limits to the in-service failure probabilities will be obtained from experimental data from the Fuel Qualification Program. Additionally, they will be calculated by TRISO fuel performance modeling tools and combined with the defective fractions characterized by fuel fabrication to evaluate the fractional release for a set of radiologically significant fission products.

The fuel performance codes will be used to calculate the failure probability of the MMR FCM TRISO fuel during normal operation, AOOs, DBAs, and BDBAs. The most limiting scenarios, in terms of expected fission product release from the FCM fuel pellet, will be modeled. The calculated fuel failure probability values will constitute predictive performance of the TRISO fuel in the MMR core. Furthermore, they will be compared to upper limits of fuel failure probability

derived from measurement on experimental data obtained through the Fuel Qualification Program.

The fission product fractional release will be calculated from these calculated in-service failure probabilities by combining them to defective fractions evaluated by characterization of the as-fabricated fuel.

5.6. Fuel Acceptance Criteria

The successful testing of the MMR FCM fuel through USNC's Fuel Qualification Program will be measured against fuel acceptance criteria. These criteria are summarized in **Table 5.2**.

Table 5.2. *MMR FCM Fuel Acceptance Criteria*

Item	Acceptance Criterion
Fuel Manufacturing	Defect fractions of the as-fabricated FCM TRISO fuel meet limits in **Table 2.2**.
Fuel Manufacturing	{{ }}
Fuel Qualification Envelope	Analysis of MMR normal operation, AOOs, DBAs, and BDBAs confirm that the FCM fuel will operate within the qualification limits presented in **Table 2.6** and **Table 2.7**.
Fuel Irradiation & Safety Testing	Fission product release fraction of FCM fuel pellet system is lower than maximum allowed value based on source term analysis showing that boundary dose limits are not exceeded.

5.7. In-service Fuel Surveillance

A fuel surveillance program will be implemented per ANSI/ANS-15.1, Sections 3.3(5) and 3.7.1(2) [87]. The fuel surveillance program will perform online monitoring to ensure that fission product release remains within operational limits.

6. CONCLUSIONS

6.1. Conclusions

The fuel qualification methodology of the MMR FCM fuel is based on international and U.S. testing experience with TRISO fuel, including the extensive irradiation testing and post-irradiation safety testing by the AGR program. The fuel qualification program is modified to account for differences in the SiC FCM fuel pellet matrix versus the traditional carbon-based matrix.

The MMR FCM fuel design specification, manufacturing, and quality controls are developed to ensure that the MMR FCM fuel exhibits exceptional performance through the use of robust fission product barrier systems. The TRISO fuel particles and FCM fuel pellets will be manufactured according to specification and tested through the fuel qualification program described in Section 5 to ensure their use in the MMR core is bounded by the FCM fuel qualification envelope. The FCM fuel qualification envelope is set to limit in-service failure of TRISO particles and FCM fuel pellets and subsequent fission product release during normal operation, AOOs, DBAs, and BDBAs.

The experimental program includes testing and characterization of unirradiated fuel and materials, fuel pellet irradiation tests in MTRs, post-irradiation safety testing of irradiated fuel pellets to measure performance in simulated accident conditions, and PIE of fuel pellets after irradiation testing and post-irradiation heat-up tests to determine fuel performance and irradiated fuel pellet material properties. **Table 6.1** summarizes the fuel qualification program.

Table 6.1. *MMR FCM Fuel Qualification Program*

Testing		Pellet Type	MTR [(a)]
Testing of Unirradiated Fuel and Materials (Section 5.1) [(b)]		{{	}}
Steady-state Irradiation Testing (Section 5.2)	{{		}}
	{{		}}
	{{		}}
	{{		}}
Short-duration Irradiation Testing (Section 5.2)	{{		}}
Post-irradiation Safety Testing (Section 5.3) [(c)]		{{	}}
Post-irradiation Examination (Section 5.4)		{{	}}

(a) {{ }}

(b) {{ }}

(c) {{ }}

The results of the fuel qualification program will be submitted through a licensing application to the U.S. NRC.

6.2. Action Requested

UNSC's fuel qualification program provides reasonable assurance that the MMR FCM fuel design can operate with a low failure rate and a level of fission product release consistent with the design basis analysis. Its results will be submitted through a licensing application to the U.S. NRC. It is requested that the U.S. NRC review Sections 5.1, 5.2, 5.3, and 5.4 and the acceptance criteria listed in Section 5.6 of this report and approve the fuel qualification methodology described therein. Following NRC approval of this fuel qualification methodology TR, it is expected that meeting the acceptance criteria in Section 5.6 of the TR by a license applicant will qualify the fuel for use as part of its application.

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