RESEARCH REACTOR

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 (FQM) for the Ultra Safe Nuclear Corporation (USNC) Micro Modular Reactor (MMR®) to be deployed at the University of Illinois Urbana-Champaign (UIUC). 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®) annular fuel pellet.

The FQM 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.

The primary objective of this TR is to provide the methodology for fuel qualification as it relates to in-reactor performance of FCM fuel pellets in the MMR core.

Safety and fission product retention for normal operations and accident conditions are achieved through a combination of a well-studied fuel form (i.e., TRISO) and a unique, additional retention barrier (i.e., FCM) to provide defense-in-depth for the containment of fission products.

In addition to the fission product barriers present in TRISO fuel, the MMR FCM fuel incorporates an additional high-integrity silicon carbide (SiC) retention barrier. The additional SiC retention 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 retention barrier, is designed to provide excellent fission product retention under MMR normal operating and transient conditions.

Because FCM is a new fuel form with limited operational experience regarding its irradiation performance, FCM fuel pellets will be tested using the fuel qualification methodology described in this TR. Activities related to USNC's fuel qualification methodology, which are discussed in this TR, include:

- Development of fuel product specifications for TRISO particles and FCM pellets (Section 2.2.2 and Section 2.2.3)
- Demonstration of pilot fuel manufacturing and quality control processes capable of consistently meeting specifications (Section 3)
- Testing and characterization of unirradiated fuel and materials (Section 6.1)
- Fuel pellet irradiation tests in material test reactors (Section 6.2)
- High-temperature safety testing of irradiated fuel pellets to measure performance in simulated accident conditions (Section 6.3)
- Post-irradiation examination of fuel pellets after irradiation testing and high-temperature safety testing to determine fuel performance of irradiated fuel pellets (Section 6.4)
- Fuel performance modeling calculations in support of fuel qualification (Section 6.5)

A fuel surveillance program (Section 6.6) will be developed and implemented per ANSI/ANS-15.1-2007 (R2018), 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 FQM serves as an input for the development of the fuel qualification plan to provide reasonable assurance that the MMR FCM fuel design is capable of operating with a low failure rate and a level of fission product release consistent with the design basis analysis. Results stemming from the execution of this FQM, and from additional fuel qualification activities, will be submitted through various reports and license applications to the U.S. Nuclear Regulatory Commission (NRC) by UIUC, and to other regulatory jurisdictions by USNC.

UIUC requests the NRC to provide a Safety Evaluation for Sections 6.1, 6.2, 6.3, 6.4, and 6.5 and the acceptance criteria listed in Section 6.7 of this report. Upon receipt of the NRC's Safety Evaluation Finding for these sections of the FQM TR, meeting the acceptance criteria in Section 6.7 of the TR qualifies the fuel for use in the UIUC MMR.

USNC intends to submit this TR to the Canadian Nuclear Safety Commission (CNSC) for joint review under the NRC/CNSC Memorandum of Cooperation dated August 2019 to promote and achieve efficiencies for planned licensing applications for its reactor technology in both countries through early regulatory engagements by familiarizing both regulators with its fuel design for reactors in the United States and Canada. The joint review will not affect the proposed applicability of the FQM TR for use by UIUC. USNC requests that the NRC continue its review and approval of the FQM TR specifically for use at UIUC as a research reactor under UIUC project number 99902094.

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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
BOI	Beginning of Irradiation
BP	Burnable Poison
[redacted]	[redacted]
CNSC	Canadian Nuclear Safety Commission
CVD	Chemical Vapor Deposition
CVI	Chemical Vapor Infiltration
DBA	Design Basis Accident
DBE	Design Basis Event
DOE	[U.S.] Department of Energy
DLOFC	Depressurized Loss of Forced Cooling
DPA	Displacements Per Atom
ECCS	Emergency Core Cooling System
EFPD	Effective Full Power Day
EFPY	Effective Full Power Year
EOI	End of Irradiation
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
FQM	Fuel Qualification Methodology
FSAR	Final Safety Analysis Report
FSV	Fort St. Vrain
GIF	Generation-IV (Gen-IV) International Forum

Abbreviation or Acronym	Definition
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 Nuclear Research 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
MWe	MW electric
MWt	MW thermal
NGNP	Next Generation Nuclear Plant
NRC	[U.S.] Nuclear Regulatory Commission
ODSL	Outer Dense Surface Layer
OPyC	Outer PyC
PDC	Principal Design Criteria
PLOFC	Pressurized Loss of Forced Cooling
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
RIA	Reactivity Insertion Accident
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

Abbreviation or Acronym	Definition
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)
UC	Uranium Carbide
UC_2	Uranium Dicarbide
UO_2	Uranium Dioxide
UIUC	University of Illinois Urbana-Champaign
UQ	Uncertainty Quantification
USNC	Ultra Safe Nuclear Corporation
2-MGEM	Two-Modulator Generalized Ellipsometry Microscope

1 INTRODUCTION

The Micro Modular Reactor (MMR®) is being developed by Ultra Safe Nuclear Corporation (USNC) to be constructed and operated at the University of Illinois 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®) annular pellets that are stacked in columns in solid hexagonal graphite fuel blocks.

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 Fuel Qualification Methodology (FQM), structured as follows:

1) Section 1.3 provides an overview of key design features of the USNC MMR
2) Section 1.4 summarizes the regulatory basis relevant to MMR fuel qualification
3) Section 2 provides the design of FCM fuel in the context of the MMR design and operating envelope
4) Section 3 provides information on FCM fuel manufacturing and quality control
5) Section 4 provides information on FCM fuel performance and failure modes
6) Section 5 provides an overview of fuel performance modeling and associated codes
7) Section 6 presents the methodology for FCM fuel qualification
8) Section 7 provides conclusions and limitations

1.2 Purpose, Scope, Regulatory Request, and Joint Review

1.2.1 Purpose

This Topical Report (TR) provides USNC's FQM, including the acceptance criteria, for qualification of the MMR fuel design to be used in the UIUC Research Reactor and future license applicants employing the USNC MMR design. UNSC's FQM serves as an input for the development of the fuel qualification plan to provide reasonable assurance that the MMR FCM fuel design will operate with a low failure rate and a level of fission product release consistent with the design basis analysis.

1.2.2 Scope

The primary objective of this TR is to provide the methodology for fuel qualification as it relates to in-reactor performance of FCM fuel pellets in the MMR core. Therefore, the TR focuses on testing that supports this objective, which includes:

- Testing and characterization of unirradiated fuel and materials (Section 6.1)
- Fuel pellet irradiation tests in material test reactors (MTRs) (Section 6.2)
- High-temperature safety testing of irradiated fuel pellets to measure performance in simulated accident conditions (Section 6.3)
- Post-irradiation examination (PIE) of fuel pellets after irradiation testing and high-temperature safety testing to determine fuel performance of irradiated fuel pellets (Section 6.4)
- Fuel performance modeling calculation in support of fuel qualification (Section 6.5)

All other aspects related to the qualification of the MMR FCM fuel are not within the scope of this TR, and will be provided in future regulatory reports, as required prior to receipt of the Operating License.

1.2.3 Regulatory Request

Results stemming from the execution of this FQM, and from additional fuel qualification activities, will be submitted through various reports and license applications to the U.S. Nuclear Regulatory Commission (NRC) by UIUC, and to other regulatory jurisdictions by USNC.

UIUC requests the NRC to provide a Safety Evaluation for Sections 6.1, 6.2, 6.3, 6.4, and 6.5 and the acceptance criteria listed in Section 6.7 of this report. Upon receipt of the NRC's Safety Evaluation Finding for these sections of the FQM TR, meeting the acceptance criteria in Section 6.7 of the TR qualifies the fuel for use in the UIUC MMR.

1.2.4 Joint Review

USNC intends to submit this TR to the Canadian Nuclear Safety Commission (CNSC) for joint review under the NRC/CNSC Memorandum of Cooperation dated August 2019 [1] promote and achieve efficiencies for planned licensing applications for its reactor technology in both countries through early regulatory engagements by familiarizing both regulators with its fuel design for USNC MMRs in the United States and Canada. The joint review will not affect the proposed applicability of the FQM TR for use by UIUC.

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 (i.e., graphite)

These features together with proper core and plant design provide the ability for HTGRs to withstand the effects of 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 [2].

1.3.2 Key Design Features of the USNC MMR

The MMR is an HTGR with a unit design power limit of 45 MWt. However, the MMR is designed for flexibility and its 100% operating limit can be decreased for specific applications by adjusting its coolant operating pressure. For UIUC, operating power level will not exceed the research reactor license limit, currently 10 MWt. The fuel qualification methodology discussed in this TR applies generically to USNC MMRs and is based on the design power limit of 45 MWt.

The MMR uses an inert gas (i.e., helium) as the heat transfer fluid. The reactor will initially be fueled with a nominal ^{235}U enrichment of 9.9% (referred to as "LEU+") in the form of TRISO particles embedded in SiC FCM pellets that are stacked in columns in solid hexagonal graphite fuel blocks.

At maximum-power steady-state conditions, the core generates 45 MW of thermal power [redacted]

[redacted] Additionally, the ^{235}U enrichment can be increased to 19.75%, without any changes to core design envelope and/or to the expected fuel temperature envelope.

The MMR is designed for passive safety response to DBAs and relies on functional containment as the primary means to limit release of radioactivity to the environment, with defense-in-depth provided by confinement through reactor systems and plant design.

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 reactor designs.

The key design features of the MMR are described below:

- The fuel is comprised of TRISO particles, which provide highly effective fission product retention capability. In response to an Electric Power Research Institute (EPRI) TR on the performance of TRISO fuel [2], the NRC issued a Safety Evaluation Report (SER) [3] [4] 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 serve as the first retention barriers 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 the 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. 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 (i.e., helium) at a pressure of approximately 6 MPa.
- 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 grade. Although it does not have nor need a leak-tested containment building, it is surrounded by a concrete structure (i.e., the citadel building) that holds the reactor vessel, the intermediate heat exchanger, and the helium circulator, and that 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 [5] 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 containment system that constitutes part of the functional containment of the MMR design. FCM fuel pellets use advanced manufacturing methods to embed TRISO particles in a SiC matrix surrounded by a SiC "Outer Dense Surface Layer" (ODSL). The ODSL provides an additional retention barrier to the release of fission products. The ODSL is made of SiC, as is one of the TRISO coating layers. [redacted] By virtue of being immediately adjacent to the helium coolant, it operates at lower temperature. Thus, the TRISO layers and the SiC ODSL constitute two separate, effective containment systems that contribute to functional containment.

The SiC used in FCM is resistant to oxidation for extended periods of time at high temperatures [6]. 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 retention 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 objective of the MMR fuel qualification plan is 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 containment systems 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: allowed research reactor maximum power (10 MWt) Power MMRs: 10-45 MWt	520 to 1400 MWe (U.S. operable reactors [7])	Full MMR power is less than decay heat of large LWR more than 24 hours after shutdown; short-lived fission product inventory is small

Feature	MMR	LWR	Remarks
Heat transfer fluid	Helium – inert gas; single phase under all conditions; low stored energy; no activation	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 retention 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 in RG 1.232 [5]
Confinement	Citadel features	N/A	Below grade vault provides fission product barrier, shielding, protection from external hazards
Safety-related alternating current (AC) power systems	None	Class 1E AC distribution and emergency diesel generators	MMR safety is provided by passive systems
Refueling frequency	[redacted]	1.5 to 2 years	[redacted]
Fuel form	Uranium oxycarbide (UCO) TRISO particles encased in FCM pellets within a hexagonal graphite fuel block	Uranium dioxide pellets encased in zirconium alloy tubes	Multiple retention barriers inherent to the fuel result in negligible fission product release from FCM in MMR during operation or accidents.; unlike LWR fuel (e.g., leaking full-length rod), the discrete TRISO fuel particles limit large release of inventory
Fuel (^{235}U) enrichment	9.9% to 19.75%	Low-enriched uranium (LEU), i.e., lower than 5%	MMR higher enrichment provides for longer core life

Feature	MMR	LWR	Remarks
Fuel damage temperature	Excessive fission product release at temperatures above 1800 ˚C for hundreds of hours; excessive oxidation at temperatures above 1800 ˚C for several hours; no particle failure at energy deposition of at least 1500 J/gUO_2	Cladding damage if exposed to steam starting at 450-500 °C for long durations; cladding ballooning and burst at 700-900 °C; hydriding and ductility loss for short times (minutes) at 1200 ˚C under design basis LOCA; cladding failure at energy deposition of 628 J/gUO_2 or less (decreasing with burnup) during reactivity Insertion accident	Zirconium-water reactions are always at play in LWRs, with the rate rapidly increasing with temperature, including breakaway oxidation in the 900-1100 °C temperature range followed by very rapid oxidation at higher temperatures; oxidation even at normal operating conditions results in hydrogen pickup and ductility loss
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	No water in the primary (He) or intermediate (molten salt) loops; high pressure (~6 MPa) of primary coolant prevents water ingress in the core	Zirconium-water reaction always produces hydrogen, and at a rapid rate if cladding temperature exceeds 1200 ˚C	Acceptance criteria limit mass of LWR fuel clad reacted
Primary system corrosion mechanisms	While helium itself is non-corrosive, oxidizing contaminants must be controlled to acceptable levels to avoid degradation of core 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 [8]. 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 NRC will be for a construction permit including a Preliminary Safety Analysis Report (PSAR) in accordance with NUREG-1537 [9]. 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 NRC.

1.4.1 Regulations Relevant to the MMR Fuel Qualification

Current 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.

As described in the UIUC TR on "Applicability of Nuclear Regulatory Commission Regulations" [8], 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.
- <u>10 CFR 50 Appendix A, "General Design Criteria"</u>: On 15 November 2023, UIUC provided a Topical Report for the MMR Principal Design Criteria (PDC) [10]to the NRC that was developed primarily based upon NRC Regulatory Guide 1.232, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors", Appendix C for Modular High-Temperature Gas-Cooled Reactors (MHTGR-DC) [5] as needed to adapt non-light-light-water reactors (non-LWR) design in lieu of direct compliance with the LWR design criteria provided in 10 CFR 50, Appendix A "General Design Criteria", and to include proposed alternatives where necessary as needed to compensate and adjust the MHTGR-DC specific to the USNC MMR design and technology.
- <u>10 CFR 50.34, "Contents of applications: technical information"</u> – 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.
- <u>10 CFR 50.34a, "Design objectives for equipment to control releases of radioactive material in effluents – nuclear power reactors"</u> – 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.
- <u>10 CFR 50.43, "Additional standards and provisions affecting Class 103 licenses and certifications for commercial power"</u> – 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.

1.4.1.1 10 CFR 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 [8].

1.4.1.2 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 [11] and RG 1.232 [5], which would allow functional containment for fission project retention rather than assuming a traditional pressure retaining containment [12]. The USNC PDC for the UIUC MMR utilizing the guidance in RG 1.232 minimizes potential gaps that may result 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.

1.4.1.3 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 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 Principal Design Criteria

10 CFR 50.34(a)(3)(i) in part, requires that an applicant for a construction permit shall include within the PSAR the PDC for the facility. On November 15, 2023, UIUC submitted a TR to the NRC titled "University of Illinois Urbana-Champaign High Temperature Gas-cooled Research Reactor: Micro Modular Reactor (MMR) Principal Design Criteria Topical Report" [10]. The PDCs applicable to the UIUC FQM TR are as follows:

- PDC 10, *Reactor Design*, provides the design criteria for reactor design and discusses the specified acceptable system radionuclide release design limits (SARRDL) for radionuclide inventory released under normal and anticipated operational occurrence (AOO) conditions as follows:

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

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 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 site boundary 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 NRC [5]. Therefore, the acceptance criteria in Table 6.3 and discussed in Section 6.7 were developed to ensure fuel performance under design basis conditions to satisfy the goals for protection of public health and safety.

- PDC 16, *Containment Design*, provides the design criteria for a functional containment design, which relies on the use of multiple retention barriers to control the release of radioactivity, as follows:

  Technical basis for FCM fuel satisfying the requirement for reactor functional containment, consisting of multiple barriers in both the TRISO fuel and FCM matrix, shall be provided. The functional containment provided by FCM will control the release of radioactivity to the environment and ensure 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 retention 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 [5].

1.4.3 NUREG-1537

For the UIUC MMR, NUREG-1537 [9] provides guidance on the conduct of licensing action reviews to 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) will 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 analysis and safety considerations, instrumentation and control features, and technical specifications requirements.

The reactor fuel section will 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 meeting these regulatory expectations is provided in the TR.

1.4.4 NUREG-0800

Although it is the standard review plan for LWRs, NUREG-0800 [13] 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 [14].

1.4.5 NUREG-2246

The 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 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 NRC issued NUREG-2246 [14].

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 3.3 and Section 3.4.

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 [14]. 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

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 NRC Policy Statements

No NRC policy statements directly apply to TRISO-coated particle fuel or to testing or monitoring of the fuel, nor does the NRC policy statement on the regulation of advanced nuclear power plants explicitly address fuel. However, many NRC activities associated with policy pertaining to source term have been identified over the years. Therefore, 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 [15], 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 [2]. More information is provided in the sections below.

Enclosure 1 of SECY-22-0008 [16] summarizes activities underway and planned by the 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” [17] 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.

1.4.6.1 Functional Containment Performance

The current LWR containment leakage requirements are provided in GDC 16 and Appendix J of 10 CFR 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 [11].

In action on SECY-93-092 [15], 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, and
- 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 [18] 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 [19], SECY-04-0103 [20], and SECY-05-006 [21], the 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 NRC instructed the staff to address the failure of the fuel particles, among other issues.

SECY-18-0096 [11] 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.

1.4.6.2 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 design basis events (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 [15], 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 [19], 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, and
- Perform source term analysis on a conservative basis to provide margin for uncertainties.

For the MMR, all 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 DOE TRISO fuel testing program (Section 1.5.3), FCM SiC has replaced carbon as the primary TRISO particle matrix material, and conservative source term analysis is planned.

1.4.6.3 NUREG-1338

In 1989, NUREG-1338 [18] documented the 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 DOE additional information and the content from two meetings held by the NRC, was completed in December 1995 [22]. 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, and
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.

1.4.6.4 Other Guidance

ANSI/ANS-53.1-2011 (R2016) [23], 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 MWe plant operated from 1967 to 1974. Reference [24] 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 MWt to achieve a net output of 330 MWe. FSV was operated from 1974 to 1989 and was licensed using a deterministic source term based on the technical information document TID-14844 [25]. The FSV fuel was highly enriched uranium/thorium carbide fissile and thorium carbide fertile TRISO particles. Two DBAs were presented in the FSV FSAR [26]: loss of forced circulation and rapid depressurization of the reactor vessel. The maximum credible accident was selected as failure of the regeneration line. The 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 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 [27].

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 [28] [29].

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 PIE were used to obtain quantitative data [30].

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 [31].

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 [31].

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 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 NRC staff and focused on adapting existing nuclear power plant regulatory requirements to the needs of NGNP licensing [32].

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 [33].

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)* [2] 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 NRC as a TR for formal review and issuance of a SER [3]. The SER concluded that there is reasonable assurance that TRISO particles produced to the 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.

Since USNC's FCM fuel form benefits from a limited operational experience regarding its irradiation performance, it will be tested through an extensive fuel qualification program. Nevertheless, the TRISO EPRI TR sets a pathway for adequate performance of TRISO fuel particles manufactured and tested under specific conditions, which establishes a strong basis for USNC's own fuel qualification program. The limitations and conditions identified by the NRC on the applicability of the TRISO EPRI TR will be satisfied by USNC's fuel qualification program, as described in Table 1.3.

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

Table 1.3. Limitations and Conditions on Applicability of TRISO EPRI Topical Report

Limitation or Condition	USNC Approach
Limitation 1 The scope of this TR applies only to the UCO TRISO particles themselves. How the final fuel form is qualified and any impacts of the fuel form or other influences of the specific reactor design beyond the fuel form on the holistic fuel performance (for instance, any uranium contamination in the compact material) is the responsibility of the vendor or designer referencing this TR.	[redacted] The TRISO particles are encased in a SiC matrix to form an FCM fuel pellet. 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 (Section 6). Manufacturing defects (i.e., exposed UCO kernel and coating layer defects) and uranium contamination (i.e., dispersed uranium), which can contribute to the release of fission products from the MMR FCM fuel, will be measured using the leach-burn-leach process (Section 3.3.2). Potential in-service failure of the FCM TRISO particles and FCM fuel pellets in the MMR core will be assessed and bounded by a combination of steady-state (Section 6.2.4) and short-duration (Section 6.2.5) irradiation testing and fuel performance modeling stress calculations (Section 6.5.1). The SiC properties required by the fuel performance models are available in open literature and will be supplemented with additional measurements (Section 6.1). [redacted]

Limitation or Condition	USNC Approach
Limitation 2 This TR applies only to UCO TRISO particles that fall within the ranges discussed in Section 5.3 of the TR. If an applicant chooses to use UO_2/UC_2 ratios or burnup values that differ meaningfully from those used in the AGR program, the applicant must provide a justification for how the burnup and carbon content ratios conform to the performance ranges discussed in Section 5.3 of the TR.	[redacted]
Condition 1 An applicant or licensee referencing this TR must evaluate any discrepancies between their fuel particles and the TRISO particles used in the AGR program – specifically, reviewing the ranges specified in Table 5-6 for stress values to capture any effects from different kernel sizes to ensure the data in the TR remain applicable.	[redacted]

Limitation or Condition	USNC Approach
Condition 2 The performance limits in Table 6-6 and Figure 6-30 of the TR are the result of different tests with distinct samples, not all of which had the maximum bounds occur during the same test. The test results include considerations for uncertainty discussed in Sections 6.4 and 6.5 of the TR. Further, when failures may occur, the data supporting the TR provides empirical evidence of failure based on aggregate test conditions rather than evidence of individual particle failure. Applicants referencing this TR must ensure that they either remain within the tested bounds or justify how their proposed operating conditions remain applicable, considering uncertainty in both the AGR test results as described in the TR and any analytical uncertainty resulting from the proposed analytical method.	[redacted] [redacted]
Condition 3 Data discussed in this TR does not consider the impacts of short-lived fission products beyond those captured in the gas phase during experiments. Any applicant or licensee referencing this TR must disposition the impacts, if any, of short-lived fission products on the safety analyses and operational dose considerations, or any other regulatory considerations resulting from short-lived fission products, in addition to the data discussed in the TR.	The impact of short-lived fission products on FCM TRISO fuel source term will be analyzed as part as a Source Term Analysis, which is outside the scope of this TR.

2 FCM FUEL DESIGN

2.1 Development of Coated Particle Fuels

A historical overview of coated particle fuel is provided in reference [34]. Significant progress on coated particle fuels began with the Dragon project [35] 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 [36].

Extensive development of TRISO fuels and understanding of TRISO fuel manufacturing and performance occurred in the 1970's and was widely published [37], 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 MWt 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 [38]. 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 [39]. 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 [40]. 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 [40].

In 2002, the 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 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 [28].
- 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 [29].
- 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 [30].
- 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 [31].

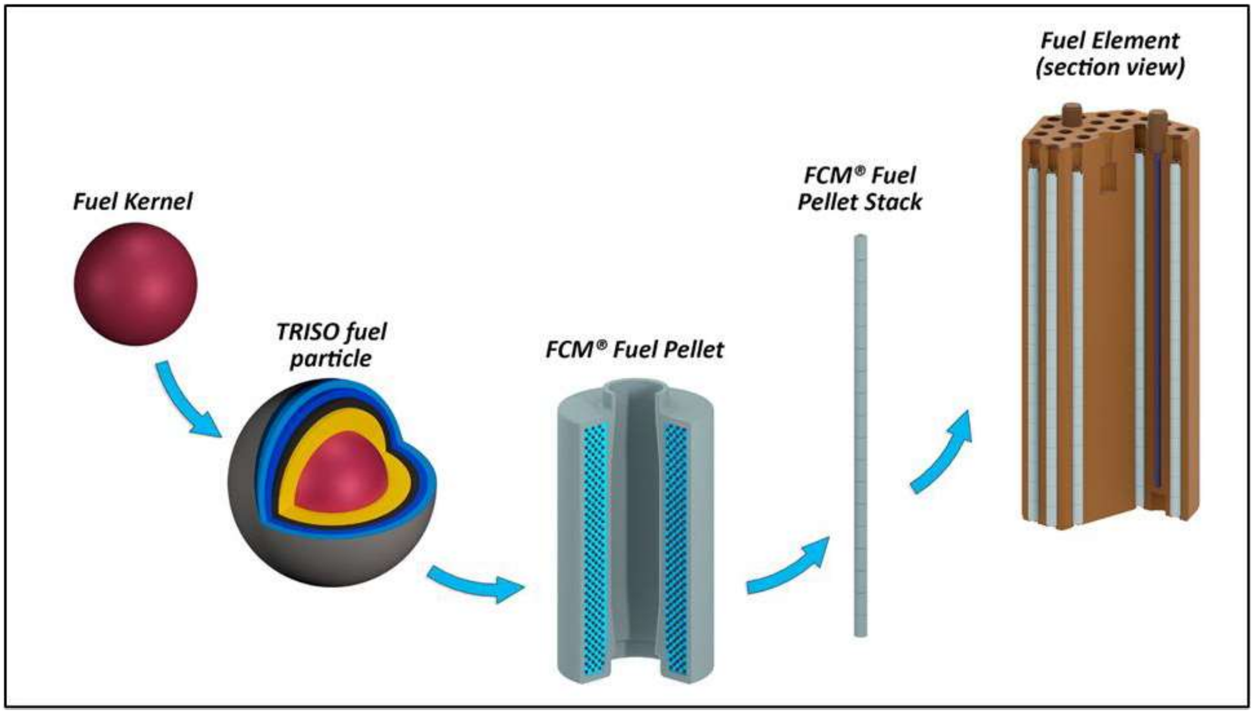
2.2 Description of FCM Fuel Design

2.2.1 Overview of MMR Fuel Design and FCM Fuel Architecture

USNC MMR fuel consists of TRISO fuel particles embedded[2] in a ceramic matrix to form an annular cylindrical FCM fuel pellet. The annular FCM fuel pellets are cooled from both the inside and outside, with coolant in direct contact with their inner and outer surfaces. FCM fuel pellets are stacked in columns and inserted into graphite fuel blocks to form fuel elements in the MMR core. A schematic of the MMR fuel design is provided in Figure 2.1.

[2] The volume ratio between TRISO particles and surrounding matrix is referred to as "packing fraction".

Figure 2.1. MMR Fuel Design



TRISO fuel particles are well developed and have known failure and fission product retention characteristics [2] [3] when manufactured to a well-defined specification, as provided in Table 5-5 of reference [2]. Systematic analyses were performed by the AGR program using the PARFUME code [41] to determine which fuel attributes are critical to fuel performance and to determine appropriate critical limits for those attributes [42].

[redacted]

The TRISO UCO fuel kernel retains a large fraction of 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. [redacted]

Conversely, in FCM pellets, replacement of the traditional carbonaceous matrix with a SiC matrix provides an additional barrier to fission product release. This additional fission product retention is mainly ensured by the SiC ODSL, which is credited in the MMR functional containment system. A fuel-free zone isolates the ODSL from the fuel region of the FCM matrix that contains the TRISO particles. Figure 2.2 shows a graphical representation of the FCM fuel architecture with the TRISO particles embedded in the FCM SiC matrix.

[redacted]

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 [43] [44] and FCM fuel [45] has been conducted by the DOE Transformational Challenge Reactor (TCR) program.

FCM TRISO particles are manufactured with procedures and equipment that are closely based upon those used for AGR fuel fabrication. Uniformity of the FCM TRISO product will be enhanced through the use of automated process control and data acquisition systems. Quality control procedures developed by the AGR program have been adopted, and adapted to MMR fuel production scale, to ensure that the level of as-fabricated particle defect fractions meets or is lower than levels established by the AGR program. 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 3.

FCM pellets are inserted into nuclear grade graphite fuel blocks to form fuel elements. The graphite fuel block is a right hexagonal prism[redacted] Fuel pellets are inserted into each of the fuel channels. [redacted] Diametral clearance allows for the dimensional change in the graphite fuel blocks and pellet stacks throughout the core life. Fuel channels double as coolant channels as FCM fuel pellets are cooled directly on their inner and outer surfaces. [redacted]

BP pellets are composed of mixtures of SiC and boron carbide (B_4C) sintered to low density. The BP pellets differ in diameter from FCM fuel pellets to ensure that FCM pellets are not inserted into BP channels. [redacted] Alignment of the elements within each fuel column is provided by a dowel and socket connection at the interface of each element in that column. The major design parameters for the graphite fuel blocks are provided in Table 2.1 and a graphical description is shown in Figure 2.3. The functions of the graphite fuel block are to maintain the geometry of the fuel and fixed BP pellets, to moderate neutrons, to channel the helium coolant through and around the FCM fuel pellets, and to maintain a coolable and controllable fuel geometry.

[redacted]. The fuel columns are sandwiched at the top and bottom by axial reflector blocks and axial shield blocks. [redacted] They contain an arrangement of coolant channels aligned with the fuel channels in the fuel blocks, and a central partial-depth cavity for handling. Above and below the upper and lower axial reflector blocks lie upper and lower axial shield blocks,[redacted]

[redacted]

[redacted]

[redacted]

[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]

[redacted]

2.2.2 FCM TRISO-Coated Particle Fuel Design

[redacted] Nominal values from the product specification of FCM TRISO fuel are compared to the AGR-1 and AGR-2 TRISO fuel properties in Table 2.2.

The TRISO kernel contains the fissile material in the form of UCO. 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. Nevertheless, in UCO kernels, the oxygen content is kept sufficient to oxidize fission products that could otherwise diffuse through the kernel and IPyC layer as high-mobility carbides to attack the SiC layer.

The porous carbon buffer layer, located between the kernel and IPyC layer, mechanically isolates the IPyC layer from kernel swelling, absorbs recoiling fission fragments, and provides a void volume to accommodate fission gases and limit buildup of internal pressure. The SiC layer acts both as a main load-bearing structural layer against internal fission gas pressure and as a 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.[redacted]

[redacted]

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 [redacted] that exhibited good performance during irradiation to a peak burnup of 10.7 %FIMA [46]. UO_2 TRISO generates CO gas in addition to gaseous fission products [47], resulting in higher internal gas pressure relative to UCO.

Table 2.2. Comparison between USNC TRISO Fuel Specification and AGR-1 and AGR-2 TRISO Fuel Properties

Particle Property	FCM TRISO	AGR-1 TRISO Baseline	AGR-2 UCO TRISO	AGR-2 UO_2
	Nominal Mean Value	Mean Measured Value	Mean Measured Value	Mean Measured Value
Kernel type	UCO	UCO	UCO	UO_2
^{235}U enrichment (wt%)	[redacted]	19.7	14.0	9.6
Kernel diameter (µm)	[redacted]	349.7	426.7	507.7
Kernel density (g/cm^3)	[redacted]	10.92	10.97	10.86
Kernel C/U (atomic ratio)	[redacted]	0.325	0.392	-
Kernel O/U (atomic ratio)	[redacted]	1.361	1.428	2.003
Kernel [C+O]/U (atomic ratio)	[redacted]	1.685	1.818	2.003
Buffer thickness (µm)	[redacted]	103.5	98.9	97.7
IPyC thickness (µm)	[redacted]	39.4	40.4	41.9
SiC thickness (µm)	[redacted]	35.3	35.2	37.5
OPyC thickness (µm)	[redacted]	41.0	43.4	45.6
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Buffer density (g/cm^3)	[redacted]	1.10	~1.04	0.99
IPyC density (g/cm^3)	[redacted]	1.904	1.89	~1.89
SiC density (g/cm^3)	[redacted]	3.208	3.197	3.200
OPyC density (g/cm^3)	[redacted]	1.907	1.907	1.884
IPyC anisotropy (BAF)	[redacted]	1.015	1.024	1.023
OPyC anisotropy (BAF)	[redacted]	1.013	1.018	1.015
Particle diameter (µm)	[redacted]	-	-	-
IPyC defect fraction	[redacted]	≤ 6.1x10^{-5}	≤ 4.8x10^{-5}	≤ 7.7x10^{-5}
SiC defect fraction	[redacted]	≤ 1.3x10^{-4}	≤ 1.2x10^{-5}	≤ 2.5x10^{-5}
Missing OPyC (defect fraction)	[redacted]	≤ 6.1x10^{-5}	≤ 9.5x10^{-4}	≤ 2.0x10^{-3}
Exposed kernel fraction	[redacted]	≤ 3.1x10^{-5}	Not reported	Not reported
Uranium contamination fraction (dispersed uranium + exposed kernel fraction)	[redacted]	-	≤ 2.5x10^{-5}	≤ 3.2x10^{-5}
Dispersed uranium fraction (U contamination fraction excluding exposed kernels)	[redacted]	3.64x10^{-7}	3.94x10^{-6}	9.66x10^{-7}

2.2.3 FCM Fuel Pellet Design

The MMR FCM fuel pellet is a SiC annular cylinder that contains the FCM TRISO particles, shown in Figure 2.4. The FCM fuel pellet consists of an inner fueled region surrounded by a fuel-free zone. The fuel-free zone isolates the TRISO particles from mechanical contact with the SiC ODSL. The TRISO particles and the ODSL provide separate and credited barriers to fission product release in the FCM fuel system.

[redacted]

[redacted]

The fuel pellet is produced using additive manufacturing (AM) using binder jet printing and chemical vapor infiltration (CVI) processes. Initially, a pre-formed annular cylindrical SiC shell with bottom and radial surfaces is printed with SiC feedstock powder. The low-density green shell is strengthened by partial infiltration of 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. [redacted]. The ODSL is designed to limit fission product release over the lifetime of the fuel core. The specifications of the FCM fuel pellet are provided in Table 2.3. [redacted]

[redacted]

[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]

[redacted]

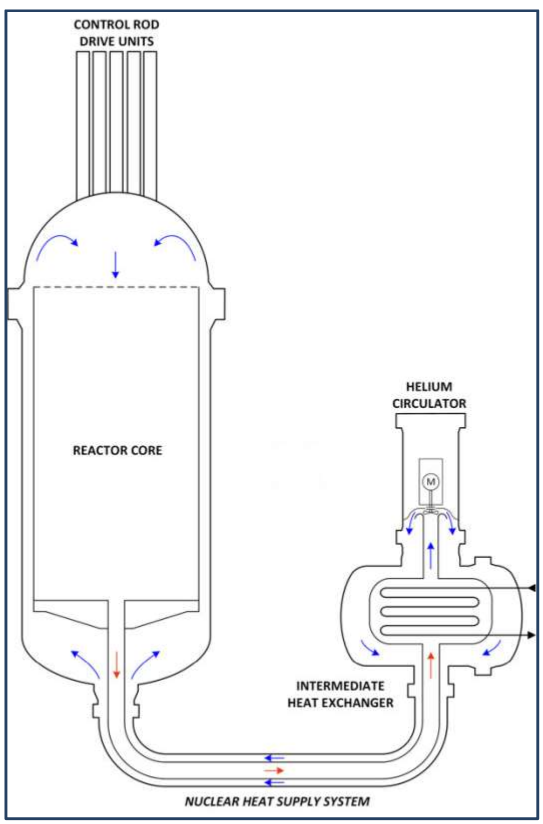
2.3 Description of MMR Core Design

The MMR is an HTGR that is designed to operate in a range of 10 to 45 MWt, but the maximum power of the UIUC MMR 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 Energy System typically consists of a Nuclear Plant and an Adjacent Plant that are physically separated. The Nuclear Plant 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 UIUC Nuclear Plant will be a single unit site. However, other sites may consist of multiple reactor units and support facilities.

The reactor core is graphite-moderated, helium-cooled, and fueled with annular 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 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 helium 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, BP, 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



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

- The helium circulator trip is a fast-acting means of reactor shutdown. It quickly drives the reactor into a subcritical, hot shutdown state from normal operation, AOOs, and DBAs, using the inherent negative temperature reactivity feedback of the fuel (Doppler).
- 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, 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.

[redacted] The [redacted] control rods are used for both reactivity control and shutdown. The layout of the RCSS is shown in Figure 2.6. All control rods in the RCSS are suspended above the core and can fully insert under gravity. Therefore, this shutdown method is not susceptible to an active failure. Furthermore, maintaining suspension of the control rods requires electrical power, so a loss of electrical power will automatically drop the rods and initiate a reactor shutdown.

[redacted]

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

The MMR core design is based on an operating power level of 45 MWt using a ^{235}U enrichment of 9.9% (referred to as "LEU+"). The MMR is designed with the flexibility to lower its operating power level to 10 MWt and/or increase its ^{235}U enrichment to 19.75%. [redacted]

2.4.1 Normal Operating Conditions

Preliminary MMR fuel operating conditions are provided in Table 2.4 for a reactor fueled with LEU+ and operating at 45 MWt power [redacted] and for a reactor operating at 10 MWt [redacted]. Table 2.4 provides core average and/or peak values for burnup, fast neutron fluence, and temperature, as well as pellet power and TRISO particle power corresponding to the reactor core power.

[redacted]

[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]

The MMR normal operating envelope is compared to AGR-1 and AGR-2 irradiation test parameters in Table 2.5 and Table 2.6. [redacted]

[redacted]

Table 2.5. Estimated MMR FCM Fuel Burnup and Fast Neutron Fluence during Normal Operating Conditions and Comparison with AGR-1 and AGR-2 Irradiation Test Conditions

Capsule	Fuel Burnup (%FIMA)			Fast Neutron Fluence ($x10^{25}$ 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	19.6	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

FCM Fuel Operating Envelope in the MMR (b)			
Peak Burnup (%FIMA)	***Peak Burnup with Uncertainty (%FIMA)***	***Peak Fast Neutron Fluence*** **($x10^{25}$ n/m^2, E > 0.18 MeV)**	***Peak Fast Neutron Fluence with Uncertainty*** **($x10^{25}$ n/m^2, E > 0.18 MeV)**
[redacted]	[redacted]	[redacted]	[redacted]

(a) Values for AGR-1 and AGR-2 are taken from Table 6-1 of reference [2].
(b) Values for USNC FCM fuel pellets are based on MMR Standard Design fueled with LEU+.

Table 2.6. Estimated MMR FCM Fuel Temperatures during Normal Operating Conditions and Comparison with AGR-1 and AGR-2 Irradiation Test Conditions

Capsule	Time-Average Minimum Temperature (°C)	Time-Average Volume-Average Temperature (°C)	Time-Average Peak 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
FCM Fuel Operating Envelope in the MMR (b)			
	Peak Fuel Temperature (°C)	**Peak Fuel Temperature with Uncertainty (°C)**	**Time-Average Peak Fuel Temperature (°C)**
	[redacted]	[redacted]	[redacted]

(a) Values for AGR-1 and AGR-2 are taken from Table 6-4 of reference [2].
(b) Values for USNC FCM fuel pellets are based on MMR Standard Design fueled with LEU+.

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 SiC matrix. Furthermore, the temperatures in Figure 2.7 are well below the melting point of the UCO kernel (> 2250 °C for the carbide phases and > 2800 °C for the oxide phase).

[redacted]

2.4.2 Transient Conditions

A preliminary assessment of transients for the 45 MWt MMR core was performed for four different scenarios:

- Pressurized loss of forced cooling (PLOFC) accident with and without scram
- Depressurized loss of forced cooling accident (DLOFC)
- Reactivity insertion accident (RIA)

At initiation of each transient, the core is assumed to be in a normal operation steady-state [redacted] (Table 2.6). The results of these transient analyses are reported in Table 2.7 for the four transient scenarios [redacted]

[redacted]

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]

Transient calculations for the 10 MWt MMR core have not been performed yet. [redacted]

2.4.3 Abnormal Conditions

Blocked Coolant Channel

Because the FCM fuel pellets are directly cooled by flowing helium, any partial fuel channel blockage could lead to a decrease in fuel cooling. [redacted]

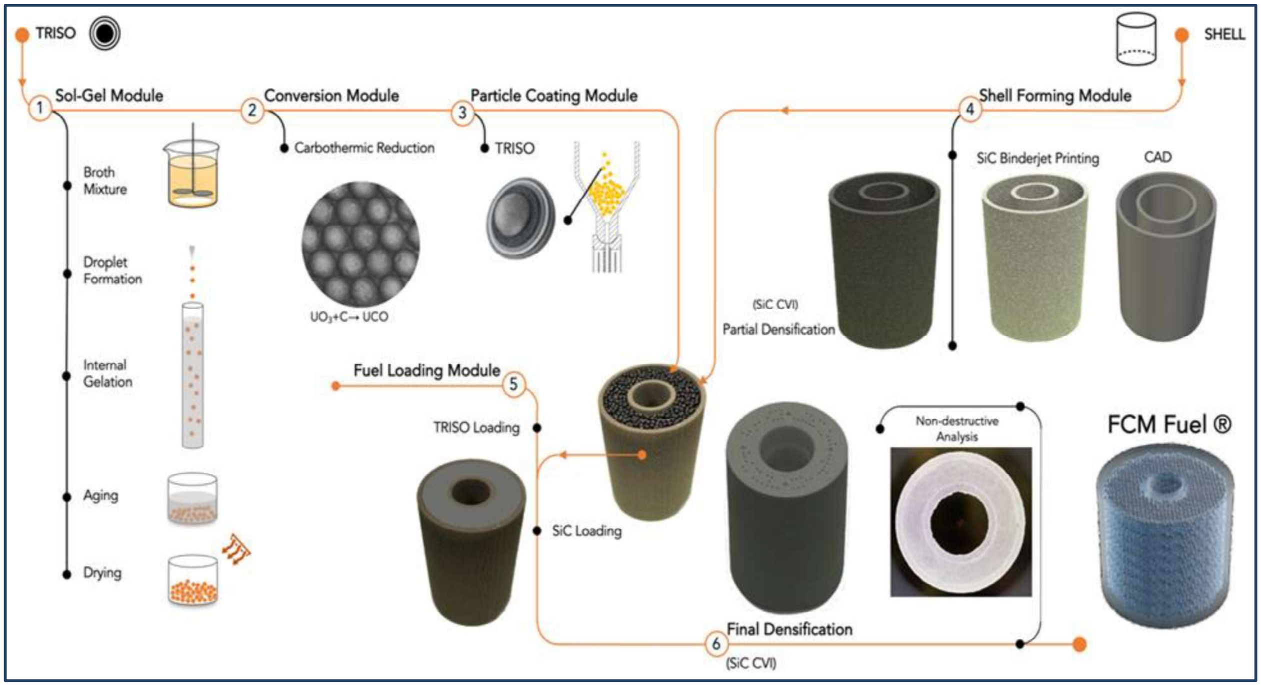
Air, Water, and Molten Salt Ingress

Risks of air, water, or molten salt (from the intermediate coolant loop) ingress into the MMR core have a very low probability of occurrence due to the high pressure of the primary coolant loop and low pressure of the molten salt heat storage system. Further evaluation will be conducted to assess these risks, and the outcome and potential impact on fuel will be addressed in a separate report.

3 FCM FUEL MANUFACTURING AND QUALITY CONTROL

FCM TRISO fuel particles are produced and inspected using fabrication equipment, fabrication procedures, and quality control procedures that closely follow those developed by the AGR program. The fabrication process for encapsulation of TRISO particles in SiC to form FCM fuel pellets was developed by DOE's TCR program. Figure 3.1 provides an overview of the FCM manufacturing process. The left side of Figure 3.1 shows the process for TRISO particle manufacturing. The right side of Figure 3.1 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 3.1. Process Flow Diagram of FCM Fuel Manufacturing Process



[redacted] AGR TRISO fuel was manufactured to a product specification that has been demonstrated to produce fuel particles with acceptable and repeatable fuel performance [2].

TRISO particles produced by the AGR program exhibited excellent performance during irradiation and safety testing [2]. Ranges of process variables provided in this section are typical of those used to manufacture AGR TRISO particles within specification limits. As experience is gained in fabrication of USNC fuel and USNC transitions to large-scale manufacturing, USNC may vary manufacturing parameters to improve FCM TRISO fuel performance, improve yield, or to ensure that TRISO particles meet the FCM TRISO fuel product specification.

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 3.1 are conducted within manufacturing modules specifically designed for each process step. The manufacturing process is scaled from pilot scale to commercial scale by adding additional manufacturing modules, operating in parallel.

3.1 Manufacturing of TRISO Particles

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 3.3) to ensure that the manufactured TRISO fuel particles meet all product specifications. [redacted]

Manufacturing of UCO Precursor Gel Spheres Using the Sol Gel process

Gel spheres are produced as precursors to UCO kernels using an internal gelation process [48] [49] [50]. Solutions of urea and hexamethylenetetramine (HMTA) are prepared according to established practice for TRISO fuel kernel manufacturing and blended in the correct proportions to form one of the two components of the sol-gel “broth”. Separately, an acid deficient uranyl nitrate solution is prepared with the required uranium-to-nitrogen ratio, uranium concentration, density, and pH by dissolution of uranium in aqueous nitric acid. Finely divided carbon is dispersed in one of the component solutions, typically along with a dispersing agent, to provide the specified carbon-to-uranium ratio in the finished kernel. The two feed solutions are chilled and blended to form the sol-gel broth. The broth is fed through vibrating needles to form droplets that are introduced into a heated silicone oil bath, where gelation takes place.

After forming, the gel spheres are aged to increase the strength of the gels, and then removed from the silicone oil. The gel spheres are washed to remove residual oil, followed by washing with ammonia water to ensure complete gelation and to remove residual reactants and soluble salts. The gel spheres are then air-dried in preparation for conversion to UCO kernels.

Conversion of Gel Spheres to UCO Kernels

Conversion is the thermal treatment process during which low-density dried gel spheres are converted to kernels conforming to specification for density, diameter, and carbon-to-uranium and oxygen-to-uranium ratios. The temperature profile during conversion is designed to produce the desired kernel properties. During thermal treatment of the dried gel spheres, uranium hydroxide hydrates are calcined [redacted] to form uranium trioxides. The calcination phase is followed by carbothermic reduction to remove oxygen and form uranium carbide [redacted] Carbothermic reduction is followed by sintering [redacted] to increase kernel density. The conversion process is carried out in inert gas or mixtures of inert gas and hydrogen.

After conversion, kernels may be subjected to an upgrading process designed to separate out irregular kernels, if the kernel manufacturing process is found to produce kernels outside of specified upper bound critical values for asphericity. [redacted] Quality control measurements (Section 3.3) are conducted after the upgrading process is completed. [redacted]

<u>Coating of UCO Kernels</u>

TRISO fuel particles are coated with multiple separate layers that are deposited by chemical vapor deposition (CVD) in series in a single fluidized bed coating system. Based on the successful German TRISO fuel development program, a continuous coating process was adopted by the AGR program for sequential deposition of the coating layers. This process has been selected by USNC for manufacturing of FCM TRISO fuel particles.

The buffer layer is first deposited from a mixture of acetylene gas diluted with argon.

The IPyC layer is deposited from a mixture of acetylene and propylene with an argon diluent. In addition to retaining fission products, the IPyC layer protects the kernel from chloride chemical attack during deposition of the SiC layer.

The SiC layer is deposited from a mixture of MTS and hydrogen or hydrogen diluted with argon. [redacted] 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 [51] [52]. 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 [53].

The OPyC layer is deposited in the same manner and with the same gas streams as the IPyC layer.

[redacted]

Deposition temperature, reactant gas flow rates, and fluidization conditions are selected to produce coating layers that meet the specifications of the individual layers. Characteristics of various parameters have different levels of tunability. The process can be controlled to achieve desired parameters such as diameter, thickness, density, and bonding between layers, while other fuel characteristics are more variable.

3.2 Manufacturing of FCM Pellets

The MMR FCM fuel form is an annular cylindrical pellet composed of TRISO particles encapsulated in a SiC matrix. The SiC pellet is coated with a SiC ODSL designed to retain fission products over the operating life of the core. The FCM manufacturing process is designed to prevent damage to TRISO particles. The mechanical compaction process used to manufacture carbon-based TRISO compacts is replaced with a three-step fabrication process:

- Fabrication of a partially densified SiC shell pre-form
- Loading of the pre-form with TRISO particles and SiC matrix powder
- Final densification of the FCM pellet using CVI

<u>Fabrication of a Partially Densified SiC Shell Pre-form</u>

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 [54]. After printing, the green shell is removed from the powder bed and excess powder is removed. The green shell is then heated to cure the binder and to increase its mechanical strength for handling. The shell wall serves as a fuel-free zone that separates the fuel matrix containing the TRISO particles from the ODSL. The shell is then partially densified using SiC CVI to bond the powder particles together. Partial infiltration increases the strength of the shell to allow for TRISO particle loading while retaining a porous shell wall. The partial densification process uses the same set of process gases used for deposition of the SiC layer on TRISO particles (MTS, hydrogen, argon) [redacted] As is the case for deposition of the TRISO SiC layer, CVI process parameters may be varied to improve the intermediate product properties.

<u>Loading of the Pre-form with TRISO Particles and SiC Matrix Powder</u>

After partial densification, the shells are filled with UCO TRISO particles and additional SiC powder. The mass of TRISO particles loaded into each shell is determined by the specified uranium content of the FCM pellet (Table 2.3). [redacted] The partially densified FCM shells are then filled with the mass of TRISO particles required to achieve the specified uranium content. SiC powder is added using a vibratory table to fill the interstices between the TRISO particles. The top portion of the FCM shell is filled with only SiC powder to form a fuel-free zone at the top of the pellet.

Final Densification of the Pellet Using CVI

Partially densified FCM shells filled with TRISO particles and SiC powder are loaded onto SiC fixtures and placed in a CVI furnace system. As for the green shell, MTS and hydrogen diluted with argon are used to infiltrate the interparticle interstices and deposit pure, stoichiometric, and crystalline SiC into the FCM matrix. As is the case for deposition of the TRISO SiC layer, process parameters may be varied to improve FCM pellet properties. [redacted]

The FCM CVI process operates at low (< 1300 °C) temperatures relative to the heat treatment process for carbon compacts (1650 – 1800 °C). The low stress and low temperature process ensures that TRISO particles are not damaged during manufacturing.

3.3 Quality Control of TRISO Particles

Intermediate and final fuel products from each batch are sampled to provide statistical information used to show adherence to the specification. Batches are combined into fuel lots and statistical quality control information is used to represent the properties of each lot.

For quality control of the FCM TRISO particles, USNC has adopted the procedures developed for quality control of AGR TRISO particles, and these procedures are being adapted to MMR fuel production scale. USNC quality control procedures ensure that critical properties of the TRISO fuel particles as identified in reference [2] are within the specified USNC tolerances.

3.3.1 Statistical Sampling Methods

Sampling methods for large-scale production (> 10^9 particles per core) are being developed based on historical experience with sampling methods used for large-scale production of TRISO fuel for the German AVR, THTR, and Modul program, in combination with modern industrial standards (e.g., ANSI sampling procedures detailed in references [55] and [56]) and methods used by the AGR program [57] to calculate confidence intervals.

Because of the large number of particles and the destructive nature of many of the quality control inspections, the entire population of TRISO particles cannot be individually inspected. Statistical sampling and analysis standards are used to determine the properties of a material lot. A lot is defined by references [55] and [56] as units of product of a single type, grade, class, or composition manufactured under essentially the same conditions. Lot size is not specified by ANSI standards. Sample size and lot acceptance criteria are based on lot size and the required inspection level. The inspection level is, in turn, defined by the stability of the manufacturing process. Selection of lot size will be informed by large scale TRISO fuel manufacturing experience gained under the German AVR program and may be adjusted as manufacturing experience is accumulated. Samples from each lot are taken using an approved random sampling procedure to ensure that the sample is representative of the TRISO particle lot. Acceptance of a TRISO particle lot is based on comparison of the attribute or variable properties of the TRISO particle sample to specified requirements. The FCM TRISO fuel specification is developed for implementation of statistical sampling methods and specifies the criteria to establish that the fuel meets the specification to a defined confidence level.

Data from inspections are recorded on electronic forms for each examination and stored on USNC's server, along with original copies of analysis results and supplier certifications. Data "books" are published for each fuel lot that summarize the results of quality control inspections and supplier certifications.

3.3.2 Quality Control of TRISO particles

USNC's TRISO particle quality control procedures were adopted, and adapted to MMR fuel production scale, from the AGR program, which refined and formalized these procedures over more than 20 years, building on German experience in large-scale TRISO fuel manufacturing. The TRISO fuel product specification focuses on the key TRISO particle properties that affect fuel performance. TRISO particle quality control procedures focus on achieving excellent and repeatable fuel performance through adherence to the TRISO fuel product specification.

The AGR program has conducted extensive TRISO fuel performance testing. This work resulted in the submittal of a TR on UCO TRISO fuel performance to the NRC by EPRI [2]. The report noted that TRISO fuel particles that exhibited property variations and were fabricated under different conditions and at different scales, exhibited remarkably similar and excellent irradiation and accident safety performance. The NRC subsequently issued a SER with the conclusion that TRISO particles produced to a TRISO fuel product specification (Table 5-5 of the TR) can be expected to perform in accordance with the TRISO fuel performance data presented in the TR [3]. The TRISO particle properties listed in Table 5-5 of the NRC-approved EPRI TR and the AGR program's quality control procedures used to measure those properties have been adopted by USNC's TRISO fuel manufacturing program.

Key properties of TRISO particles inspected and controlled by USNC procedures are summarized in Table 3.1.

Table 3.1. Key Properties of TRISO Particles Subject to Quality Control Procedures

Property	Kernel	Buffer	IPyC	SiC	OPyC	TRISO	[redacted]
Kernel C/U and O/U ratios	X						
Impurities	X						
Diameter	X					X	
Density	X	X	X	X	X		
Layer thickness		X	X	X	X		[redacted]
Microstructure				X			
Anisotropy			X		X		
Aspect ratio	X				X		
Layer defect fraction			X	X	X		
Exposed kernel fraction						X	
Dispersed uranium						X	
Average uranium content						X	

Quality Control of UCO Precursor Gel Spheres

Uranium feedstock is analyzed for uranium content, isotopic content, and impurities per USNC's specification for uranium feedstock. Urea, HMTA, carbon, aqueous ammonia solutions, and silicone oil are received from approved suppliers with certificates of analysis. Purified water is used for formation of gel spheres to ensure that extraneous impurities are not introduced above specified levels.

Finished gel spheres have low mechanical strength and are subject to internal quality control measures to ensure suitability for input into the conversion process that results in UCO kernels. Quality control measurements to ensure adherence to the FCM TRISO fuel specification are performed after conversion to UCO kernels.

Quality Control of UCO Kernels

Conversion of gel spheres forms UCO kernels with sufficient strength to be handled for measurement and metallographic examination. UCO kernel diameter, aspect ratio, density, and carbon-to-uranium and oxygen-to-uranium ratios are measured. Kernel lots that do not meet specification are rejected and recycled.

[redacted]

[redacted]

[redacted]

[redacted]

Quality Control of TRISO-coated Particles

Because the TRISO particle coating layers constitute the first containment system against fission product release, they are subject to a series of focused measurements to ensure that as-fabricated TRISO particles meet the specification required to achieve desired performance.

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

3.4 Quality Control of FCM Pellets

Quality control processes for FCM pellets are currently being developed to ensure the in-service integrity of the pellet and to allow the pellet ODSL to be credited as a retention barrier to fission product release.

Critical attributes of FCM pellets inspected and controlled by USNC procedures are:

- FCM pellet dimensions
- Thickness of the ODSL
- Leak-tightness of the ODSL
- Mechanical integrity of the ODSL
- Uranium loading

Measurement of FCM pellet outer dimensions ensures that the FCM pellets do not mechanically interact with the graphite fuel block in a manner that causes damage to the graphite block or to the FCM pellets [redacted] All FCM pellets are weighed and dimensionally measured using an automated inspection system with resolution sufficient to ensure that specified tolerances are met.

The ODSL provides a retention barrier to the diffusion of fission products released by the TRISO particles contained in the FCM pellet matrix. The thickness of the ODSL impacts the fractional fission product release rate for a given temperature and operating time. [redacted]

[redacted]

[redacted]

4 FCM FUEL PERFORMANCE

The MMR FCM fuel form is an annular cylindrical pellet composed of TRISO particles held in a SiC matrix and encapsulated in a high density SiC shell. FCM fuel is designed and fabricated such that the TRISO particles and the ODSL component of the FCM fuel pellet act as complementary barriers to fission product release. In particular, FCM fuel:

- Incorporates a TRISO particle design adapted from the designs of the AGR-1 and AGR-2 TRISO particles that demonstrated high fission product retention during tests exposing them to temperatures higher than expected for MMR normal operating and transient conditions.
- Surrounds the TRISO particles with a partially densified SiC matrix that protects them from mechanical damage.
- Includes a fuel-free zone that isolates the TRISO particles from the ODSL.
- Encapsulates the SiC matrix and fuel-free zone in a SiC ODSL that provides an additional barrier to fission product release.
  - The ODSL [redacted] operates at a lower temperature. Although theoretically susceptible to the same degradation reactions as the TRISO SiC layer, the ODSL lower operating temperature and separation from the TRISO particles make it effectively diverse. Additionally, the SiC matrix geometrically separates the TRISO particles from the ODSL and acts as an internal getter mitigating any potential chemical degradation at its interface with the ODSL.
  - The ODSL retains radioactivity from fission of trace uranium impurities outside the TRISO SiC layer. This “tramp” or “dispersed” uranium is the source of much of the release of fission products from traditional TRISO fuel compacts early during irradiation.

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 [2]. 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 verifying (1) the performance of TRISO particles embedded in the FCM fuel pellet, (2) that the geometry of the FCM pellet is maintained during operation, and (3) that the FCM pellet provides an additional barrier to fission product release. TRISO-coated particle fuel performance is discussed below.

4.1 TRISO Particle Performance

Traditional 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 TRISO 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 [59]. 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. However, PyC anisotropy is limited by specification to values demonstrated to provide good overall TRISO fuel 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 reversal of the strain depends on the density and degree of anisotropy of the PyC layer (controlled by 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 more 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 TRISO 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 TRISO SiC layer. The irradiation behavior of the outer coating layers is shown schematically in Figure 4.1 and summarized in Table 4.1 [60]. An illustrative representation of the corresponding stress in the IPyC and SiC layers as a function of fast neutron fluence is shown in Figure 4.2.

[redacted]

Figure 4.1. Irradiation Behavior of Outer Coating Layers in a TRISO Particle

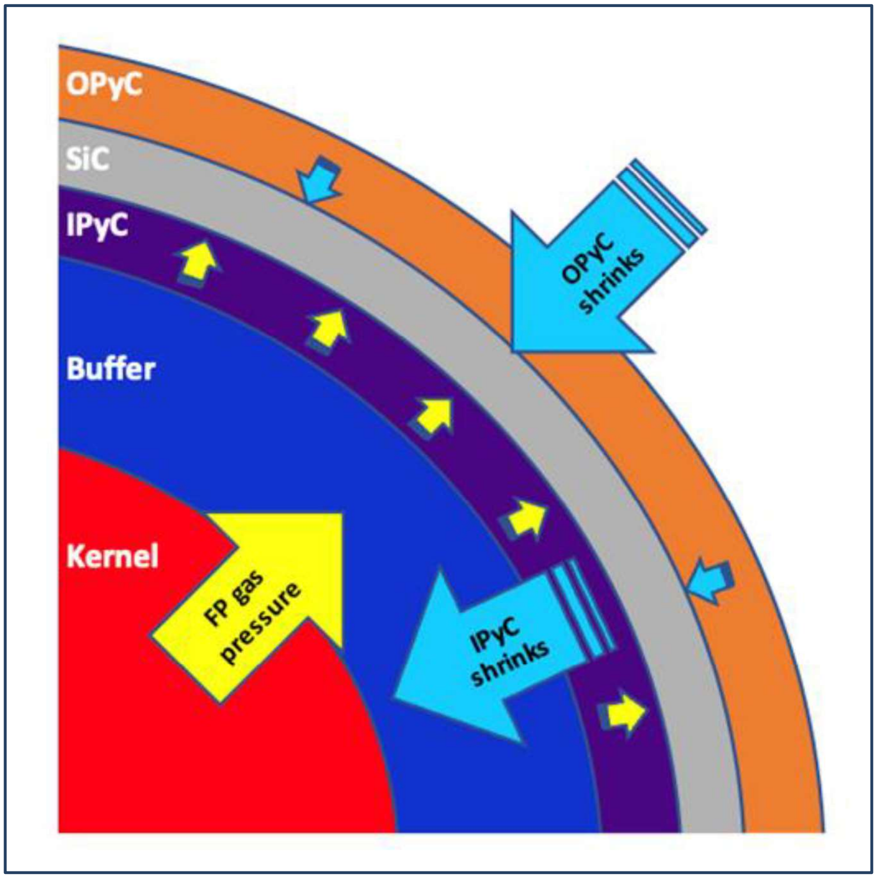
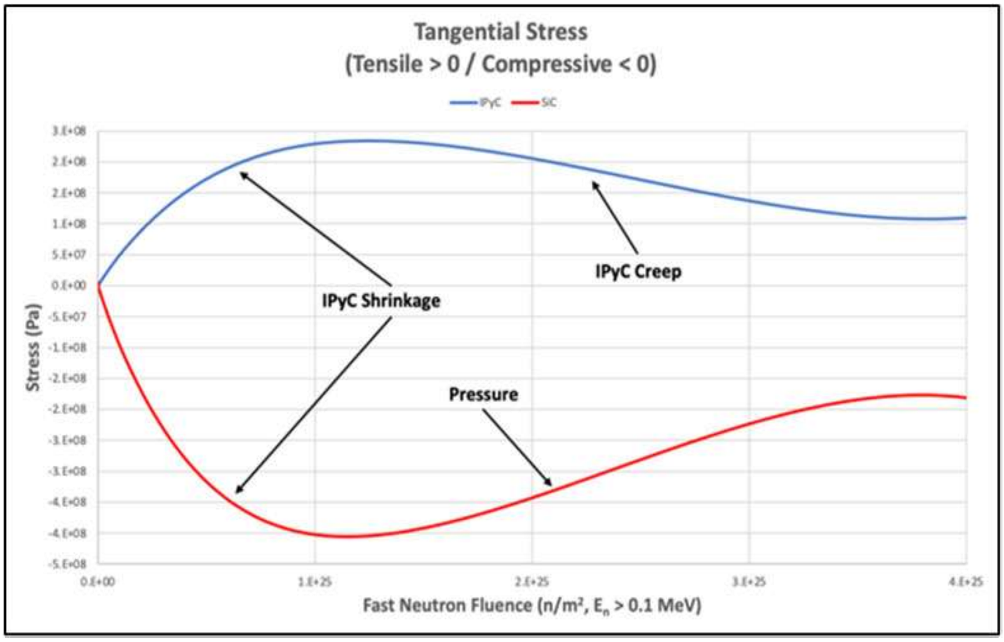


Table 4.1. Irradiation Behavior of TRISO Particle Components

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 4.2. 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 coating layers
- Fabrication defects
- 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 layer provides the primary retention barrier to fission product release in the TRISO particle, 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, fission products from dispersed uranium are expected to be a substantial portion of the radioactivity released from TRISO fuel early during irradiation. Limits are prescribed in the TRISO particle specification to control release by this mechanism, but the SiC ODSL of the FCM pellet should substantially reduce this source.

4.2 TRISO Particle Failure Mechanisms

Based on historical irradiation experience, the following failure mechanisms of TRISO-coated fuel particles have been identified [61] [2]:

- 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 (e.g., noble metals or lanthanides) or CO 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

Two types of failure can result from these failure mechanisms: SiC failure and TRISO failure; SiC failure is defined as the loss of fission product retention capability by the SiC layer, but it leaves at least one of the PyC layers intact; TRISO failure, also referred to as "full failure", corresponds to the loss of leak-tightness of all three outer coating layers (i.e., IPyC, SiC, and OPyC), opening a direct pathway to fission product release from the kernel into the SiC matrix.

Additionally, potential OPyC cracking due to interaction with the surrounding FCM pellet matrix is discussed in Section 4.2.4 [redacted]

4.2.1 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 (Figure 4.2).

<u>Production of Carbon Monoxide</u>

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 [61]. 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 [46]. 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

[redacted]

A so-called "SiC tensile stress metric" is described in reference [2] 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.

[redacted]

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. Rather, a TRISO fuel performance modeling code calculation of the stress level in the SiC layer would provide a direct comparison between the FCM TRISO fuel and the AGR-1 and AGR-2 fuel designs at their relevant operating conditions. [redacted]

4.2.2 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 4.2.3) 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 layer can then lead to its failure.

Cracking of the IPyC layer, and potential subsequent SiC failure, tends to occur early during irradiation when tensile shrinkage stresses are maximum (Figure 4.2).

4.2.3 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 [62].

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 (Figure 4.2).

4.2.4 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 [63] [46]. [redacted]

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 compact 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 [46]. The presence of the gap may prevent OPyC cracks from propagating into the SiC layer by being disconnected at the debonded SiC/OPyC interface.

4.2.5 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 phase 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 [63] [46].

4.2.6 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 [61] [2].

For instance, palladium (Pd) 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 observed Pd release from TRISO particles through intact SiC, but no corrosion or attack of SiC was observed on these as-irradiated particles [2]. 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 [63]. However, AGR-2 PIE identified shallow isolated regions of SiC degradation with no signs of cracked or degraded IPyC [46]. 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). [redacted]

[redacted]

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 [64]. 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. [redacted] Lanthanide oxides have lower mobility and increased retention in the kernel [64].

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, which facilitates 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 [2]. [redacted]

4.2.7 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 [65]. 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 and irradiation durations of 620 and 559 effective full power days (EFPD), respectively.

Safety testing of AGR-1 and AGR-2 fuel for ~300 hours at temperatures of 1800 °C also did not provide any evidence of SiC decomposition. [redacted]

4.2.8 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 [63] [46].

As noted in reference [63], 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 [46].

[redacted]

[redacted]

4.3 FCM Pellet Performance

The purpose of the FCM fuel pellet is to:

- Provide an additional barrier to release of fission products to the coolant
- Isolate TRISO particles from chemical interaction with coolant impurities and from mechanical interaction with the graphite fuel block
- Transfer heat from the TRISO particles to the coolant

As discussed in Section 2.2, an MMR FCM fuel pellet is composed of TRISO particles embedded in a porous SiC matrix, a higher-density fuel-free zone surrounding the fuel region, and a SiC ODSL at near theoretical density. The FCM pellet structure serves to isolate the TRISO particles from chemical interaction with coolant impurities, with which the FCM pellet is in direct contact, and from mechanical interaction with the graphite fuel block in case of excessive combined dimensional changes during irradiation. The fuel-free zone facilitates manufacturing and prevents mechanical interactions of the ODSL with the TRISO particles. The ODSL [redacted] functions as an additional barrier to fission product release from FCM fuel.

The performance of the FCM pellet during normal operation or under transient scenarios is mainly driven by stress that develops under the influence of irradiation-induced dimensional changes (i.e., swelling) and thermal gradients at high power.

The pellet thermal gradient is a function of pellet power density, pellet surface temperature, and pellet matrix thermal conductivity. In the annular FCM pellet, the designed balance in cooling between the inner and outer surfaces minimizes temperature gradients and subsequent thermal stresses. The elastic modulus, Poisson's ratio, and coefficient of thermal expansion determine the stress state in the pellet in response to the temperature gradient. The elastic modulus is a function of pellet density but it does not depend strongly on fast neutron fluence or temperatures within the range of MMR operating conditions. Thermal conductivity is a function of pellet density, fast neutron fluence, and temperature. Prior to irradiation, the thermal conductivity of the printed SiC is dependent on orientation, with lower values in the radial (XY) direction. This orientation dependence disappears during irradiation [43].

The FCM pellets are densified and the ODSL applied in a continuous process, as is the case for TRISO layers. While TRISO SiC layers are coated by CVD (i.e., deposition on the IPyC outer surface), densification of the SiC matrix and ODSL of the FCM pellet is performed by CVI (i.e., deposition within the structure of the SiC matrix). 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. The differences in processing conditions may affect the grain size of the deposited SiC, but do not affect its purity, stoichiometry, or crystallinity. These attributes are critical for attaining predictable swelling behavior [67] and for preventing degradation in strength [68] from the effects of irradiation displacement damage. CVD SiC mechanical and thermal properties and their dependence on irradiation are reported in reference [69]. 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. Specifically, under neutron flux, the swelling behavior of CVD SiC varies across three temperature regimes [69] [70]:

- 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.

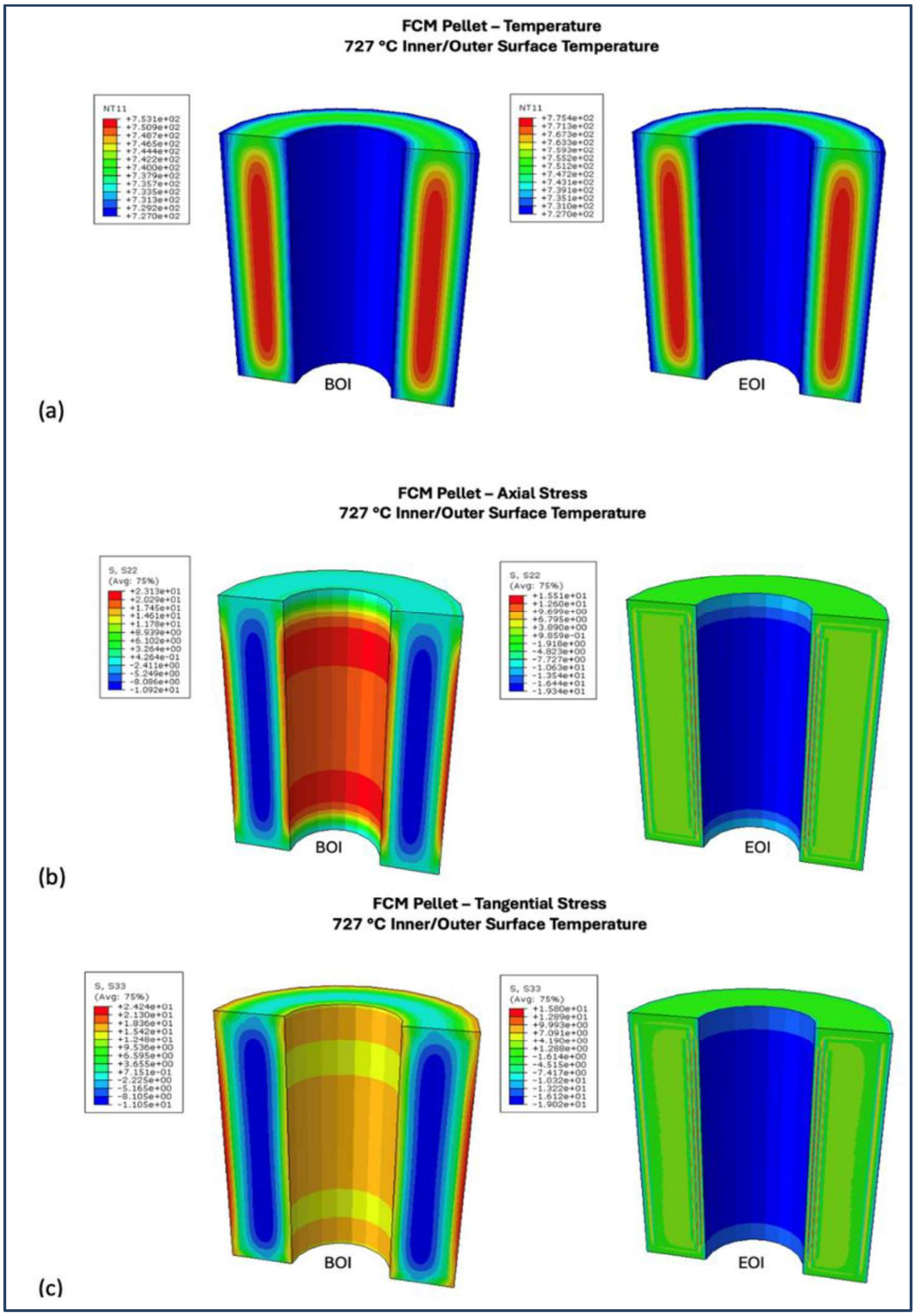
Thermal and mechanical properties of the FCM CVI SiC matrix (e.g., characteristic strength and Weibull modulus – that are used to calculate failure probability using Weibull analysis [71], or thermal conductivity) were measured on unirradiated material and material irradiated to 2.3 displacements per atom (dpa) [43]. 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 (Section 6.1).

The SiC matrix and ODSL provide thermal resistance between the reactor coolant and TRISO particles. Although unirradiated CVI SiC has relatively high thermal conductivity, it degrades with fast neutron fluence [43]. As reported in reference [72], in the point defect regime, the thermal conductivity of irradiated CVD SiC decreases and saturates at relatively low dose (a few dpa). 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 [73].

During irradiation, fission heat generation from the embedded TRISO particles creates a thermal gradient between the inner and outer surfaces of the FCM pellet. The stress in the ODSL results from the balance between the differential thermal expansion and swelling rate of the ODSL and matrix. As previously discussed, irradiation-induced swelling in SiC is a function of temperature and fast neutron fluence. As fast neutron fluence accumulates, the volume of the lower temperature near-surface region of the fuel pellet increases at a faster rate relative to the higher temperature of its interior region. The increased low temperature swelling in the inner and outer regions acts to relieve the tensile stress near the pellet surface. As differential swelling proceeds, the stress state of the inner and outer regions transitions from tensile stress to compressive stress.

An illustration of this behavior is shown in Figure 4.3. The example assumes an FCM pellet with inner and outer surfaces cooled at 1000 K (727 °C), as shown in Figure 4.3 (a). During initial reactor startup, 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 generates tensile stress on the pellet inner and outer surfaces. Left plots of Figure 4.3 (b) and Figure 4.3 (c) show that the maximum axial and tangential stresses at the beginning of irradiation (BOI) are tensile and near the pellet inner and outer surfaces. As irradiation proceeds and fast neutron fluence is accumulated, the colder inner and outer surfaces of the FCM pellet swell more than the hotter interior region. The increased volume of the inner and outer surfaces counteracts the larger thermal expansion of the interior matrix region. Swelling rapidly reaches saturation, but the differential swelling between the ODSL and the matrix (ODSL swelling is larger than matrix swelling) remains larger than their differential thermal expansion (matrix thermal expansion is larger than ODSL thermal expansion). Right plots of Figure 4.3 (b) and Figure 4.3 (c) show the maximum axial and tangential stresses at the end of irradiation (EOI), after saturation of swelling. The stresses on the inner and outer surfaces become compressive while the stresses in the interior region of the matrix become slightly tensile.

Figure 4.3. Illustration of the Evolution of the Stress in FCM Pellets as a Function of Accumulated Fast Neutron Fluence



4.4 FCM Pellet Failure Mechanisms

The potential failure mechanisms specific to irradiation of the FCM pellet include:

- ODSL degradation caused by reactor operation
- FCM pellet fracture
- Chemical attack of the ODSL
- Corrosion of the ODSL by impurities in the reactor coolant
- Thermal decomposition of the ODSL or SiC matrix at high temperatures

An FCM pellet failure is defined as the loss of leak-tightness to fission product release by the ODSL.

4.4.1 ODSL Degradation Caused by Reactor Operation

The SiC ODSL surrounds the fuel-free zone in the FCM pellet. The fuel-free zone acts as a supporting substrate for the ODSL, preventing fracture from the hydrostatic stress applied by the pressurized helium coolant in the MMR core.

Because the FCM pellet matrix is assumed to be non-retentive of fission products, cracks or holes in the ODSL provide a direct path for release of fission products from the FCM pellet. The release through a compromised ODSL would be limited to that present in the FCM pellet matrix from defective or failed TRISO particles, from dispersed uranium present outside of leak-tight TRISO SiC layers, and from diffusive release through intact TRISO particle layers.

ODSL integrity during manufacturing is ensured by pressure testing at levels in excess of MMR core coolant pressures under worst case conditions, by visual inspection, and by dimensional inspection (Section 6.1).

SiC is among the hardest of ceramic materials, with a measured Vickers hardness ranging from 20.7 – 24.5 GPa [67]. [redacted]

4.4.2 FCM Pellet Fracture

As seen in Figure 4.3, the maximum axial and tangential stresses in the FCM pellet are tensile and near its inner and outer surfaces at BOI and become compressive as irradiation progresses and irradiation-induced swelling counteracts the effects of thermal expansion.

The early-irradiation tensile stress at the inner and outer surfaces of the FCM pellet can lead to cracking of the ODSL if it becomes larger than the fracture strength of the ODSL. A cracking of the ODSL results in a loss of leak-tightness to fission product release.

As indicated in Figure 4.3, FCM pellets are most vulnerable to fracture at BOI, when the fission product inventory is low. The response of FCM pellets to well-defined and representative temperature gradients will be addressed through testing and analysis (Section 6.2.5 and Section 6.5.1).

Similarly, Figure 4.3 shows that the inner region of the FCM matrix that surrounds the TRISO particles is subjected to increasing tensile stress as irradiation progresses. Cracks resulting from levels of tensile stress above the fracture strength of the SiC matrix have been shown to propagate into embedded TRISO particles and cause TRISO failure [74]. [redacted]

4.4.3 Chemical Attack of the ODSL

Similar to potential chemical degradation of the SiC layer in TRISO particles, noble metals, lanthanides, and/or rare earth elements that migrate through the SiC matrix after release from TRISO particles may attack the ODSL and challenge its leak-tightness to fission product release. Additionally, CO released from exposed or failed TRISO particles could also challenge the integrity of the ODSL although CO production in UCO fuel is very limited.

Chemical attack of the ODSL is similar to SiC layer degradation (Section 4.2.6) but with a thicker SiC layer operating at lower temperature. Consequently, the risk of degradation of the ODSL is lower than the risk for the TRISO SiC layer. Furthermore, the ODSL is separated from the source of fission products by the SiC matrix, including a fuel-free zone, which is subject to the same risk of chemical attack. A potential chemical attack by fission products would affect the SiC matrix before the ODSL, further reducing the risk of its degradation.

4.4.4 Corrosion of the ODSL by Impurities in the Reactor Coolant

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 [75]. Because of the large surface area of graphite in HTGR cores, gaseous oxygen exists at very low levels in HTGR cores.

Oxidation of the FCM fuel pellets could occur if exposed to these impurities at high temperatures, and the FCM SiC matrix and SiC ODSL could slowly oxidize over the lifetime of the FCM fuel in the MMR core.

Oxidation of SiC may occur in passive and active regimes. Passive oxidation of SiC results in the formation of a protective silica (SiO_2) surface layer that slows diffusion of oxygen to the surface. During active oxidation, on the other hand, ambient oxygen reacts with SiC to form volatile SiO and CO. Active oxidation occurs if ambient conditions (e.g., very low oxygen levels and high temperatures) do not allow the formation of a protective SiO_2 surface layer.

Oxidation testing of CVD SiC [76] [77] [78] and 3-D printed SiC [79] was carried out at various temperatures and levels of impurities in steam and/or air. For CVD SiC, analysis indicated a transition from passive to active oxidation near 1300 °C at low oxygen levels (2 ppm), and a measured mass loss during active oxidation near zero below 1450 °C for oxygen levels up to 1000 ppm [77]. For 3-D printed SiC, oxidation tests were conducted in both flowing steam and air environments at 1300 °C and 1425 °C for isothermal hold times of 100 hours, and both tests indicated passive oxidation of the tested SiC reaction tubes [79]. [redacted]

Furthermore, the impact of coolant impurities on ODSL and TRISO performance is mitigated by the corrosion resistance of SiC, the absence of water in the MMR coolant loop, and the MMR coolant cleanup system. The MMR utilizes two systems to establish and maintain low impurity levels in the helium coolant:

- A Primary Coolant Cleanup System whose function 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 purging the vessel system before the final filling with high-purity helium.
- A Helium Purification System whose function is to remove fission products, 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 and returned in a separate line to the intermediate heat exchanger.

In the specific case of a fractured ODSL, TRISO particles could also get oxidized if exposed to coolant impurities at high temperatures. Oxidation of the OPyC layer (or IPyC layer if exposed by defective, failed, or missing SiC and OPyC layers) could mobilize fission products into the SiC matrix. The SiC coating layer in the TRISO particle is expected to be more resistant to oxidation than PyC but it could slowly oxidize over the lifetime of the FCM fuel in the MMR core. Exposed kernels (from as-fabricated defects or in-service failures) are vulnerable to hydrolysis, which could increase fission gas release.

4.4.5 Thermal Decomposition of the ODSL or SiC Matrix at High Temperatures

The issue of thermal decomposition of the SiC in the matrix or ODSL is similar to the issue of thermal decomposition of the SiC layer in the TRISO particle (Section 4.2.6), with the ODSL exposed to temperatures equal to or lower than TRISO particle temperatures.

Reference [80] notes that SiC covered by a layer of PyC decomposes more slowly than exposed SiC. Consequently, the decomposition rate for the ODSL could be faster than the decomposition rate for the SiC layer in the TRISO particle. [redacted]

4.5 Fission Product Transport in FCM Fuel

The various radioisotopes created by fission are mainly in two atomic mass groups (i.e., 90-100 and 130-140) and include over 30 elements. The specific radioisotopes have different nuclear (i.e., amount produced, half-life, energy of emitted radiation, neutron cross-section), physical (i.e., gas or solid), and chemical (i.e., 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.

4.5.1 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 [81]. 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.

4.5.2 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 [82]. 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. However, the fundamental knowledge of all transport phenomena is limited, 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 – [redacted] The slow rate of diffusion through the ODSL results in significant fission product decay and in subsequent reduced radiological release.

The influence of the SiC microstructure on the effective diffusivities in the SiC layer was evaluated in reference [83]. Electron backscatter diffraction was used to show that layer formation by 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. AGR PIE compared measured fission product release from coarse grain, high-aspect ratio SiC microstructures to fine grain equiaxed microstructures [83]. 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 [83].

4.5.3 Fission Product Transport in FCM Pellet

The MMR FCM fuel uses SiC in two different forms to prevent fission product release from the FCM fuel pellet: the TRISO SiC layer acts as the primary barrier to fission product release from TRISO particles; the ODSL provides an additional barrier on each FCM fuel pellet to retain fission products released by TRISO particles into the SiC matrix in which they are embedded. This approach provides defense-in-depth to functional containment.

The fission product release behavior of SiC has been characterized by multiple international programs [84]. 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 [41] to experimentally measured fission product release was performed using data from the AGR-1 irradiation test [85]. The AGR-1 test was irradiated to a peak burnup of 19.6 %FIMA and a maximum fast neutron fluence of 4.3×10^{25} n/m^2 (E > 0.18 MeV) over 620 EFPD over a range of time-average compact temperatures spanning from 800 to 1200 °C [28]. 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 over-predicted at low temperature and high burnup and under-predicted at high temperature and low burnup.

[redacted]

[redacted]

5 FCM FUEL PERFORMANCE MODELING

Fuel performance modeling is used by USNC to:

- Predict fuel performance during MMR normal operating and transient conditions to support fuel design, fuel fabrication, fuel optimization, and safety evaluation
- Predict fuel performance in MTRs for design of irradiation experiments that support fuel qualification
- Assist post-irradiation analysis of fuel behavior in MTRs in support of fuel qualification

The two figures of merit for evaluation of FCM fuel performance are in-service probability of failure (i.e., loss of leak-tightness to fission product release) and fission product fractional release (i.e., the ratio between the amounts of fission products released from the TRISO particles or FCM pellets and produced in the TRISO fuel kernels and dispersed uranium).

For fuel qualification purposes (Section 6), analysis of MMR FCM fuel performance (i.e., TRISO particles and FCM pellets) will be conducted through a combination of modeling (Section 6.5), using validated and approved fuel performance modeling codes, and reliance on testing data (Section 6.2.4, Section 6.2.5, and Section 6.3).

TRISO particle and FCM pellet performance modeling will be performed using TP3-DIFFUSION and Abaqus. TP3-DIFFUSION is being developed by USNC using models and calculation methods similar to the PARFUME code that is used by the AGR program as its basis for TRISO fuel performance modeling [41]. Abaqus [86] was chosen by USNC as its reference commercial finite element analysis package.

Specifically, TP3-DIFFUSION will be used to calculate the potential release of fission products from the TRISO particles into the FCM SiC matrix, through the ODSL and, subsequently, into the coolant; Abaqus will be used to predict the thermomechanical stress on the FCM fuel pellet to assess the structural integrity of the ODSL and its probability of failure.

Both codes will use established TRISO and FCM performance models (i.e., material properties and physical models) in combination with specific experimental data (e.g., material properties) measured by USNC's fuel qualification program (Section 6.1). Both codes will be adequately validated prior to being used for modeling and simulation supporting fuel licensing activities.

Verification and validation for TP3-DIFFUSION and Abaqus, fuel performance modeling methodologies for TRISO particles and FCM pellets, and uncertainty quantification associated with fuel performance calculations will be addressed in separate reports.

5.1 TP3-DIFFUSION

TP3-DIFFUSION is used to model radionuclide transport through TRISO particles and FCM fuel pellets and subsequent fission product release to the coolant.

TP3-DIFFUSION is a 1D spherical symmetric diffusion model that can be configured for any number of concentric spherical layers, each assigned with different properties. Additionally, the model allows the application of multiplication factors to the surface areas and volumes of the outermost layer to account for the different surface area to volume ratio for cylindrical (or other) geometries. This is used to properly model the ODSL, whose geometry is cylindrical.

TP3-DIFFUSION relies on a finite difference scheme for calculation of the diffusion of atomic species. The governing equations are solved as a set of matrices and concentrations are calculated from the resulting tridiagonal systems of equations by using the Thomas algorithm.

The diffusion module contains three different metrics to calculate fission product release:

- The ratio of release rate to birth rate, commonly referred to as "R/B ratio", is used to calculate the ratio of release rate to production rate of short-lived radioactive isotopes, which release rate can reach a near-equilibrium level during the FCM fuel reactor core residency.
- Total fractional release is used to calculate the ratio between the total release of long-lived radioactive isotopes up to a given point in time and their total production up to that point in time.
- Decayed fractional release is used to calculate the ratio between the total decayed release of long-lived decaying radioactive isotopes up to a given point in time and their decayed total production up that point in time; this represents the ratio between the remaining released inventory after allowing for decay and the remaining total produced inventory after allowing for decay.

The MMR FCM fuel pellets are annular cylindrical pellets containing randomly distributed spherical TRISO fuel particles. Diffusion and release of fission products from the TRISO particle kernels into the FCM pellet are modeled by:

- Using concentric spherical material layers to represent the TRISO particle constituents (kernel, buffer, IPyC, SiC, OPyC, [redacted]
- Adding two concentric material layers to represent the SiC matrix and the ODSL:
  - The concentric SiC matrix is lower density SiC with high diffusivity for radioactive species of interest, and therefore is assumed not to provide any retention barrier to radionuclide transport.
  - The thin high-density ODSL, with low diffusivity based on currently available SiC diffusion coefficients, acts as the last retention barrier in the FCM fuel system.

TP3-DIFFUSION includes a model to account for fission product recoil from the TRISO particle kernel into its buffer. Direct recoil into the buffer can occur when fission products are produced at distances from the buffer smaller than their mean ranges in the kernel and with momenta directed towards the buffer. In the case of direct recoil, a portion of the source term gets allocated from the kernel to the buffer.

Meshing for the FCM fuel model is done for each layer individually. Finer meshing can be used at the layer interfaces, where large differences in diffusivities can exist and potentially cause numerical instabilities. A coarser mesh is typically used in the central regions of the layers.

Additionally, only one node is used to represent failed layers, which are assumed non-retentive of fission products, to reduce mass balance errors.

In each layer of the FCM fuel model, a material type is assigned and material properties (i.e., thermal conductivity and diffusion coefficients) are specified. TP3-DIFFUSION does not include thermomechanical or irradiation-induced properties (except for Cs diffusivity in SiC). The low power density of the MMR core [redacted] results in a temperature gradient in the TRISO particle that is negligible in the evaluation of fission product transport.

5.2 Abaqus

Abaqus is a commercial general-purpose finite element-based software package developed and marketed to provide engineering solutions and multiphysics simulation software for product design, testing, operation, and analysis [86].

For MMR FCM fuel, Abaqus will be relied on to perform stress calculations in the FCM fuel pellet. Stress calculations on the FCM fuel pellet allow to determine the probability of mechanical failure of the ODSL as a function of power density, temperature, and thermal gradient across the pellet for various pellet geometries. Stress in the FCM fuel pellet is impacted by the material properties of the SiC matrix and ODSL – such as elastic properties, irradiation-induced swelling, thermal properties, or Weibull parameters – which are included in the performance model.

6 METHODOLOGY FOR FUEL QUALIFICATION

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

The methodology for fuel qualification, as it relates to in-reactor performance, is intended to describe the irradiation testing activities conducted on FCM fuel pellets to ensure that they will be tested under irradiation conditions that bound their expected performance in MMR cores during normal operation and transients. Results from irradiation testing will be used to demonstrate that the FCM fuel can be safely operated in MMR cores.

FCM fuel design relies on the previously demonstrated TRISO particle architecture [2] [3] and on the FCM pellet as retention barriers to fission product release. [redacted]

Activities related to USNC's fuel qualification program, which are discussed in this TR, include:

- Development of fuel product specifications for TRISO particles and FCM pellets (Section 2.2.2 and Section 2.2.3)
- Demonstration of pilot fuel manufacturing and quality control processes capable of consistently meeting specifications (Section 3)
- Testing and characterization of unirradiated fuel and materials (Section 6.1)
- Fuel pellet irradiation tests in MTRs (Section 6.2)
  - Steady-state irradiation testing primarily aimed at providing performance demonstration of integral FCM fuel pellets under irradiation conditions that bound those of the MMR core (Section 6.2.4)
  - Short-duration irradiation testing specifically aimed at establishing FCM pellet failure threshold as function of mechanical stress induced by thermal gradient and SiC swelling (Section 6.2.5)
- High-temperature safety testing of irradiated fuel pellets to measure performance in simulated accident conditions (Section 6.3)
- PIE of fuel pellets after irradiation testing and high-temperature safety testing to determine fuel performance of irradiated fuel pellets (Section 6.4)
- Fuel performance modeling calculations in support of fuel qualification (Section 6.5)

6.1 Testing and Characterization of Unirradiated Fuel and Materials

[redacted]

[redacted]

- [redacted]
- [redacted]
- [redacted]
- [redacted]
- [redacted]
- [redacted]
- [redacted]
- [redacted]

[redacted]

6.2 Fuel Pellet Irradiation Tests in Material Test Reactors

FCM fuel consists of two separate containment systems that retain fission products. The net fission product release from FCM fuel pellets is the product of the release fractions from the TRISO particles and the release through the ODSL. [redacted]

[redacted]

- [redacted]
- [redacted]
- [redacted]

6.2.1 Types of Irradiation Tests

Two types of irradiation tests will be used to meet the three irradiation testing objectives:

1) Steady-state, long-term irradiation testing (months to years) will be used to test the integral performance of FCM fuel pellets under conditions that bound MMR operating conditions; the steady-state irradiation testing will be conducted at the High Flux Reactor (HFR) in Petten, the Netherlands.
2) Short-duration irradiation testing (minutes to hours) will be used to provide information on the failure threshold of FCM pellets operating at high power densities; the short-duration irradiation testing will be conducted at the Massachusetts Institute of Technology Nuclear Research Reactor (MITR) in Cambridge, MA.

6.2.2 Irradiation Test Articles

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
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[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]

6.2.3 Irradiation Testing Parameters

Key parameters controlled during irradiation testing are:

- TRISO particle power density – TRISO particle power density is primarily a function of the fission rate density within the UCO kernel; TRISO particle power determines the temperature gradient across the TRISO particle radius, the fuel pellet power density, and the temperature gradient across the fuel pellet.
- FCM pellet peak temperature – Fission product release through TRISO SiC and PyC layers is a function of temperature; FCM pellet operating temperatures determine the diffusive fission product release from TRISO particles.
- FCM pellet surface temperature – FCM fuel pellets are manufactured with a SiC ODSL as a retention barrier to fission product release; diffusive fission product release through the ODSL is a function of ODSL temperature.
- Pellet temperature gradient – Because SiC is a ceramic with low ductility, large thermal gradients may result in stresses in the FCM pellets beyond the FCM failure strength; during steady-state operation, pellet thermal gradients are correlated to TRISO particle power and pellet thermal conductivity; reactor power ramps may also result in thermal gradients.
- TRISO particle burnup – Burnup is used as a measure of the cumulative number of fissions within the TRISO kernels; the associated effects of fission (e.g., kernel swelling and build-up of internal fission gas pressure in the TRISO particles due to fission gas) are generally correlated as a function of burnup.
- Fast neutron fluence – Swelling in the SiC FCM pellet matrix, the ODSL, and the SiC coating layers in TRISO particles is a function of fast neutron fluence; dimensional

changes and creep in the TRISO buffer, IPyC, and OPyC layers are also a function of fast neutron fluence.

6.2.4 Steady-state Irradiation Testing

[redacted] Steady-state irradiation testing will be conducted at the Petten HFR, in the Netherlands.

Target values for key parameters of the steady-state irradiation testing of scaled-down pellets are provided in Table 6.2. These parameters are selected to bound FCM fuel temperatures, temperature gradients, burnup, and fast neutron fluence during MMR normal operation and AOOs. In particular, the target scaled-down pellet power is defined so that the temperature gradients and resulting stresses in the SiC matrix and ODSL experienced by the FCM pellets in the MMR core are bounded by irradiation testing.

[redacted]

[redacted]

[redacted]

[redacted]	[redacted]	[redacted]
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[redacted]	[redacted]	[redacted]
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[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]

[redacted] [redacted]

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[redacted]

6.2.5 Short-duration Irradiation Testing

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

6.3 High-temperature Safety Testing

[redacted]

[redacted]

[redacted]

[redacted]

6.4 Post-irradiation Examination

PIE [redacted] from irradiation testing and high-temperature safety testing will be used to confirm the irradiation performance of FCM fuel pellets and embedded TRISO fuel particles. [redacted]

[redacted]

6.5 Fuel Performance Modeling in Support of Fuel Qualification

Fuel performance modeling will be used to assess the performance of the MMR FCM fuel based on experimental results from steady-state irradiation testing, short-duration irradiation testing, and high-temperature safety testing.

[redacted]

6.5.1 ODSL Stress and Failure Probability

[redacted]

[redacted]

[redacted]

6.5.2 ODSL Diffusivity and Fission Product Release

The dense ODSL on the FCM pellet provides retention of fission products and is consequently credited in the MMR functional containment. Its thickness is designed to provide an additional fission product barrier [redacted]

[redacted]

6.6 In-service Fuel Surveillance

A fuel surveillance program will be implemented per ANSI/ANS-15.1-2007 (R2018), Sections 3.3(5) and 3.7.1(2) [93]. The fuel surveillance program will perform online monitoring of the reactor coolant for abnormal fission product activity during startup, startup testing initial operations, and full power equilibrium operation to ensure that fission product release remains within operational limits.

In particular, online monitoring of fission gas release from the FCM fuel will serve as an indicator of TRISO particle and FCM pellet failure. Anomalous fuel performance resulting in excessive fission gas release will be readily detectable and actionable to ensure no undue risk to public health and safety.

6.7 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 6.3.

Table 6.3. 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	The thickness of the as-fabricated FCM ODSL meets limit in Table 2.3. FCM pellets have zero defective ODSL (i.e., 100% of FCM pellets pass the hermeticity test).
Fuel Qualification Envelope	Analysis of MMR normal operation, AOOs, and DBAs confirm that the FCM fuel will operate within the qualification limits presented in Table 2.5 and Table 2.6.
Fuel Irradiation & Safety Testing	Fission product release fraction of FCM fuel pellet during MMR normal operation, AOOs, and DBAs is lower than maximum allowed value based on source term analysis showing that boundary dose limits are not exceeded.

7 CONCLUSIONS

7.1 Summary of Fuel Qualification Methodology

The FQM 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 AGR program.

The primary objective of this TR is to provide the methodology for fuel qualification as it relates to in-reactor performance of FCM fuel pellets in the MMR core.

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 6 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, and DBAs.

The experimental program includes testing and characterization of unirradiated fuel and materials, fuel pellet irradiation tests in MTRs, high-temperature safety testing of irradiated fuel pellets to measure performance in simulated accident conditions, and PIE of fuel pellets after irradiation testing and high-temperature safety testing to determine fuel performance. Table 7.1 summarizes the fuel qualification methodology. Additionally, Table 7.2 shows how irradiation testing will bound normal operating and transient conditions in MMR cores.

[redacted]

[redacted]	[redacted]		[redacted]	[redacted]
[redacted]	[redacted]		[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
	[redacted]	[redacted]	[redacted]	[redacted]

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[redacted]

7.2 Action Requested

UNSC's FQM serves as an input for the development of the fuel qualification plan to provide reasonable assurance that the MMR FCM fuel design is capable of operating with a low failure rate and a level of fission product release consistent with the design basis analysis. Results stemming from the execution of this FQM, and from additional fuel qualification activities, will be submitted through various reports and license applications to the NRC by UIUC, and to other regulatory jurisdictions by USNC.

UIUC requests the NRC to provide a Safety Evaluation for Sections 6.1, 6.2, 6.3, 6.4, and 6.5 and the acceptance criteria listed in Section 6.7 of this report. Upon receipt of the NRC's Safety Evaluation Finding for these sections of the FQM TR, meeting the acceptance criteria in Section 6.7 of the TR qualifies the fuel for use in the UIUC MMR.

7.3 Limitations

The use of this FQM for the MMR FCM TRISO fuel is subject to the following limitations:

- Application of the FQM is limited to tests described in Section 6.1, Section 6.2, and Section 6.3 with the objective of establishing a qualification envelope for in-reactor performance of FCM fuel pellets in the MMR core.
- The design of the FCM fuel pellets and embedded TRISO particles corresponds to the description provided in Section 2.2.3 and Section 2.2.2, respectively, and meets the same specifications (Table 2.3 and Table 2.2) that are used for fuel qualification testing.
- The MMR FCM TRISO fuel will be used in MMR cores corresponding to the description provided in Section 2.3 and will perform under conditions bounded by the normal operating conditions detailed in Section 2.4.1 and transient conditions detailed in Section 2.4.2.
- The MMR fuel operating envelope presented in Table 2.5 and Table 2.6 will be detailed in safety analysis reports submitted through license applications to the NRC; it will be demonstrated that the MMR fuel operating envelope is bounded by the FCM fuel qualification envelope, which will be defined by planning of fuel irradiation testing (Section 6.2.4 and Section 6.2.5) and high-temperature safety testing (Section 6.3) as part of the execution of the fuel qualification methodology described in this TR.
- Flow blockage of coolant through the fuel channels will be demonstrated to result in a negligible impact on the performance of the MMR FCM TRISO fuel; it will be demonstrated that any potential impact is bounded by the FCM fuel qualification envelope.

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