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Project No. 99902071

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Submission of X Energy, LLC (X-energy) Xe-100 Topical Report: TRISO-X Pebble Fuel Qualification Methodology, Revision 2

REFERENCE: Letter from M. van Staden to Nuclear Regulatory Commission dated 30 April 2021, "Submission of X Energy, LLC (X-energy) Xe-100 Topical Report: TRISO-X Pebble Fuel Qualification Methodology"

On April 30, 2021, X Energy, LLC (X-energy), submitted Revision 1 of the above referenced topical report. The purpose of this letter is to submit Revision 2 of the topical report and is intended to supersede previous revisions. This revision includes an updated figure based on NRC staff feedback during a meeting on June 4, 2021, and provides both proprietary and non-proprietary versions of the report. It is provided for NRC review and approval as indicated in the report and is expected to be referenced in future Xe-100 licensing applications. The specific review schedule will continue to be developed with X-energy's NRC project manager; however, we request that acceptance review and schedule planning occur within 30 days of commencement and a review duration of 12 months be considered.

This report contains commercially sensitive, proprietary information and, as such, we are requesting that this information be withheld from public disclosure in accordance with 10 CFR 2.390, "Public inspections, exemptions, request for withholding," paragraph (a)(4). Additionally, certain information in this report was determined to contain Export Controlled Information (ECI). This information must be protected from disclosure pursuant to 10 CFR 810. Enclosure 1 is the Non-Public version of the report which contains non-redacted sensitive information. This proprietary information is appropriately marked, and an affidavit providing the basis for this request is provided in Enclosure 2. A redacted copy of the report that contains non-proprietary content is provided in Enclosure 3.

If you have any questions or require additional information, please contact Ingrid Nordby at inordby@x-energy.com.

Sincerely,

Travis Chapman
U.S. Licensing, Xe-100 Program
X Energy, LLC

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Enclosures:
1) Xe-100 Topical Report: TRISO-X Pebble Fuel Qualification Methodology (Proprietary)
2) Affidavit
3) Xe-100 Topical Report: TRISO-X Pebble Fuel Qualification Methodology (Non-Proprietary)

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Enclosure 1
X Energy, LLC Xe-100 Topical Report: TRISO-X Pebble Fuel Qualification Methodology
(Proprietary)

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Enclosure 2
Affidavit

X Energy, LLC
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Affidavit Supporting Request for Withholding from Public Disclosure (10 CFR 2.390)

I, Martin van Staden, Vice President, Xe-100 Program Manager, of X Energy, LLC (X-energy) do hereby affirm and state:

1. I am authorized to execute this affidavit on behalf of X-energy. I am further authorized to review information submitted to or discussed with the Nuclear Regulatory Commission (NRC) and apply for the withholding of information from disclosure. The purpose of this affidavit is to provide the information required by 10 CFR 2.390(b) in support of X-energy's request for proprietary treatment of certain commercial information submitted in Enclosure 1 to X-energy's letter 2021-XE-NRC-012 from T. Chapman to the NRC which provides topical report XE00-R-R1ZZ-RDZZ-L-000633 that provides a description of the fuel qualification methodology for TRISO-X fuel for X Energy, LLC's Xe-100 reactor.

2. I have knowledge of the criteria used by X-energy in designating information as sensitive, proprietary, confidential, and export-controlled.

3. Pursuant to the provision of paragraph (b)(4) of 10 CFR 2.390, the following is furnished for consideration by the NRC in determining whether the information sought to be withheld from public disclosure should be withheld.

a. The information sought to be withheld from public disclosure in Enclosure 1 is owned by X-energy. This information was prepared with the explicit understanding that the information itself would be treated as proprietary and confidential and has been held in confidence by X-energy.

b. The information sought to be protected in Enclosure 1 is not available to the public.

c. The information contained in Enclosure 1 is of the type that is customarily held in confidence by X-energy, and there is a rational basis for doing so. The information X-energy is requesting to be withheld from public disclosure includes technical information related to the design, analysis and operations associated with our Xe-100 high-temperature, gas-cooled, pebble bed advanced reactor design that directly impact our business development and commercialization efforts. X-energy limits access to this proprietary and confidential information in order to maintain confidentiality.

d. Enclosure 1 contains information about the planned activities of X-energy related to the development of the Xe-100 and TRISO-X fuel design bases, forecast design development timeframes, and relate to the commercialization strategy for our Xe-100 advanced reactor. Public disclosure of the information contained in Enclosure 1 would create substantial harm to X-energy because it would reveal valuable technical information regarding X-energy's design development, competitive expectations, assumptions, current position, and strategy. Its use by a competitor could substantially improve the competitor's position in the design, manufacture, licensing, construction, and operation of a similar competing product.

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e. Additionally, Enclosure 1 is assessed to contain certain information that is considered Export Controlled Information (ECI) under the provisions of 10 CFR 810. I have personal knowledge of the criteria used by X-energy to evaluate documents for ECI and affirm that this information must be withheld from public disclosure.

f. The Proprietary Information contained in Enclosure 1 is transmitted to the NRC in confidence and under the provisions of 10 CFR 2.390; it is to be received in confidence by the NRC. The information is properly marked.

I declare under the penalty of perjury that the foregoing is true and correct. Executed on September 2, 2021.

_____

Dr. Martin van Staden

Vice President, Xe-100 Program Manager

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Enclosure 3
X Energy, LLC Xe-100 Topical Report: TRISO-X Pebble Fuel Qualification Methodology
(Non-Proprietary)

Xe-100 Topical Report

TRISO-X Pebble Fuel Qualification Methodology

Configuration Classification	:	**XE00-R-R1ZZ-RDZZ-L**
Revision	:	**2**
Status	:	**Approved**
Issue Date	:	**16-Aug-2021**
Project	:	**Xe-100**

ELECTRONIC SIGNATURES: DOCUMENT APPROVAL

Preparer:
Electronically signed by Paul Loza
16-Aug-2021 19:56

Reviewer:
Electronically signed by Eben Mulder
31-Aug-2021 06:47

Reviewer:
Electronically signed by Travis Chapman
31-Aug-2021 06:47

Approver:
Electronically signed by Martin Van Staden
31-Aug-2021 12:51

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10 CFR 810 Export-Controlled Information Disclaimer

Department of Energy Acknowledgement and Disclaimer

This material is based upon work supported by the Department of Energy under Award Number DE-NE0008472.

EXECUTIVE SUMMARY

X Energy, LLC (X-energy) is developing the Xe-100 reactor for deployment in multiple markets, with an emphasis on the United States and Canada as initial entry points. This report summarizes the planned fuel qualification approach and methodology for the pebble fuel form used in the Xe-100, which includes supplemental testing of TRI-structural ISOtropic (TRISO)-coated Uranium Oxycarbide (UCO) fuel particles in spheres. The objectives of the supplemental testing are to (1) provide linkage to the Department of Energy (DOE) Advanced Gas Reactor (AGR) Program data base for TRISO-coated UCO fuel particles, (2) provide data for TRISO-coated UCO fuel particles in spheres with a formal Nuclear Quality Assurance (NQA-1) pedigree, especially for code validation, and (3) demonstrate that TRISO-coated UCO fuel particles in spheres fabricated in the U.S. perform in a manner consistent with user and regulatory requirements.

The Xe-100 design is based on more than fifty years of extensive experimental and commercial HTGR operations and significant design activity by HTGR reactor suppliers and several governments. Pebble bed HTGR technology dates to the late 1960s. China currently has a modular HTGR plant with two 250 MWt pebble bed reactors in cold commissioning, and the HTR-10 experimental pebble bed plant in operation since 2003. The reactor core in a pebble bed HTGR is composed of hundreds of thousands of 60 mm diameter spherical fuel elements (also referred to as pebbles) that form the pebble bed. Pebble bed cores are continuously refueled on-line, with the pebbles added at the top of the reactor, withdrawn from the bottom, and returned to the top of the reactor for further irradiation until the pebbles achieve full burnup of the fuel. Each Xe-100 pebble contains thousands of ~1 mm diameter TRISO-coated fuel particles. The fuel particles consist of a fissionable fuel kernel surrounded by coating layers of pyrolytic graphite and silicon carbide for retention of radionuclides (RNs).

Building upon the extensive investigations of TRISO-coated fuel particle performance beginning in the 1970s and the substantial progress made by the more recent Advanced Gas Reactor (AGR) Fuel Development and Qualification Program, the Xe-100 design has adopted the low enriched (LEU) UCO TRISO-coated particle being tested and qualified by the AGR Program as the reference design. This design selection takes advantage of the demonstrated superior high-temperature performance and high burnup capabilities of the UCO particle compared to the uranium oxide (UO_2) particle that was used in the design of earlier pebble bed modular HTGRs, as discussed in Section 5.3.

This report provides information related to TRISO-coated particle fuel and fuel pebbles for the following purposes:

- Identify existing regulations, regulatory guidance, and licensing precedents relevant to the qualification of fuel for the Xe-100 design;
- Summarize existing understanding, data, and analysis methods regarding TRISO-coated particle fuel performance;
- Review reactor and fuel design and performance requirements;
- Describe the national and international experience base that provides the technical foundation for the fuel qualification approach;
- Describe planned fuel fabrication, irradiation, safety testing activities, and approach to qualify the reference fuel (UCO TRISO-coated particles); and

- Provide a documented basis for engagement with regulators on the planned approach and information required for qualifying the Xe-100 reference fuel.

"Fuel qualification" as used in this report refers to the following program elements:

- Establishment of fuel product, equipment, and feedstock specifications;
- Implementation of a fuel fabrication process capable of consistently and reliably meeting the specifications at the required scale;
- Implementation of statistical QC/QA procedures to demonstrate that the product specifications have been met;
- Irradiation of statistically sufficient quantities of fuel with monitoring of in-pile performance and post-irradiation examination to demonstrate that normal operation performance requirements are met; and
- Post-irradiation heating tests (safety testing) of statistically sufficient quantities of irradiated fuel to demonstrate that accident condition performance requirements are met.

For spherical fuel elements, there are additional fuel qualification tasks associated with their various physical properties. For example, the spheres are tested to confirm that their structural integrity is maintained through repeated cycling through the core and within the fuel handling and storage systems. The methods for confirming structural integrity and other important physical properties are briefly summarized in this report. Future licensing applications to the U.S. Nuclear Regulatory Commission (NRC) will include the Xe-100 fuel specifications/requirements upon which the reactor analyses are based.

The preliminary as-manufactured fuel quality requirements and in-service performance requirements for prismatic modular HTGR fuel for the Next Generation Nuclear Plant (NGNP) are given in

Table S-1. In this table heavy metal (HM) contamination is defined as exposed uranium fraction outside any coating layer and reported as a kernel equivalent per particle count. [[

]]^{P} A fuel product specification for the Xe-100 reference fuel was prepared that is consistent with the requirements in

Table S-1. [[

]]^{P} The national and international experience base in fabrication of UCO fuel particles and in fabrication of spherical fuel elements containing both UO2 and UCO fuel particles implies that UCO TRISO-coated particles in spheres can be successfully mass produced to this specification and that this fuel will meet in-service performance requirements with high confidence.

Table S-1: Preliminary Xe-100 Fuel Specification

Parameter	Max. Expected	Design
As-Manufactured Fuel Quality:		
HM contamination	[[ ]]^{P}	$\leq 2.0 \times 10^{-5}$
Missing or defective buffer	[[ ]]^{P}	$\leq 2.0 \times 10^{-5}$
Missing or defective IPyC	[[ ]]^{P}	$\leq 1.0 \times 10^{-4}$
Defective SiC	[[ ]]^{P}	$\leq 1.0 \times 10^{-4}$
Missing or defective OPyC	[[ ]]^{P}	0.02
In-Service Fuel Failure:		
Normal operation	[[ ]]^{P}	$\leq 2.0 \times 10^{-4}$
Core heat-up accidents	[[ ]]^{P}	$\leq 6.0 \times 10^{-4}$

Table S-2: Supplemental Test Program to Qualify UCO in Spheres

[[ ]]^{P}	[[ ]]^{P}	[[ ]]^{P}
[[ ]]^{P}	[[ ]]^{P}	[[ ]]^{P}
[[ ]]^{P}	[[ ]]^{P}	•[[ ]]^{P}

The AGR Program is focused upon qualifying UCO TRISO-coated particles in cylindrical fuel compacts that are characteristic of prismatic fuel elements. The process conditions for making fuel spheres are different from those for making fuel compacts. In particular, the pressures for pressing spheres are higher than the pressures for molding fuel compacts (typically, 300 MPa vs. less than 20 MPa). This difference is, however, somewhat offset by the lower packing fraction of the particles in the sphere volume (~10 to 30%) compared with that of particles in the cylindrical compact volume (~35 to 45%). The lower packing fraction results in less likelihood of contact between the outer surfaces of particles. X-energy will demonstrate that these fabrication process differences will not have any deleterious effects on the UCO TRISO-coated particles (a low probability concern since both high enriched uranium (HEU) UCO and LEU UO_2 TRISO-coated particles in spheres have performed well). In addition, new equipment has been constructed for manufacturing fuel spheres in the U.S. by X-energy's wholly-owned subsidiary, TRISO-X. [[

]]^P

[[

]]^P

This report presents background information on the design of the Xe-100 reactor plant and the safety design approach. A review of applicable NRC regulations and guidance (and related precedents), relative to fuel qualification, that must be satisfied for licensing the Xe-100 in the U.S. is presented.

The report presents a review of historic experience in testing and qualification of TRISO-coated fuel particles for both prismatic and pebble bed modular HTGRs. Information on the successful accomplishments to date of the AGR Fuel Development and Qualification Program, [[the results of which will be built upon by the Xe-100 supplemental Fuel Qualification Program,]]^P is presented. The report describes the approach that X-energy has adopted to incorporate and leverage the experience and results of the AGR Program and to demonstrate that UCO TRISO-coated fuel particles in spheres will meet the fuel quality and performance requirements for the Xe-100 design. The approach described for fuel qualification is applicable to all future deployments of the Xe-100 irrespective of licensing application type (i.e., Part 50 Operating License, Part 52 Combined Licenses or Design Approval and/or Certification, and future prospective Part 53). Collectively, these activities for the methodology X-energy is pursuing to qualify the TRISO-coated particle fuel and fuel pebble described herein for use in the Xe-100 reactor design.

[[

]]^P

The X-energy Fuel Qualification Program's objective is to provide reasonable assurance of public health and safety throughout the lifecycle of the Xe-100 reactor, and that the excellent performance of U.S.-produced UCO TRISO-coated particle fuel will be maintained through the controls described herein. This report describes several future activities, including on-going fuel irradiation and testing in the AGR program, X-energy's design-specific fuel irradiation and testing, and in-reactor pre-operational and commissioning tests. The resulting information from these activities will be provided in future revisions to this licensing topical report and/or in the design and licensing bases submitted for site or design-specific licensing applications under 10 CFR 50, 10 CFR 52, or future 10 CFR 53.

X-energy requests that the NRC staff review and approve the methodology and acceptance criteria as described in Section 7 of this topical report.

CONFIGURATION CONTROL

Document Change History

Rev.	Date	Preparer	Changes
A	4-Sep-2020	Travis Chapman	Initial document developed from previous white paper content.
1	28-Apr-2021	Paul Loza	Issue as Licensing Topical Report
2	16-Aug-2021	Paul Loza	Updated to include full withholding mark-up and update to fuel performance envelope (Figure 25).

Document Approval

Action	Designation	Name	Signature	Date
Preparer	Senior Licensing Engineer	P Loza		
Reviewer	SVP, Chief Scientist	E Mulder		
Concurrer	Lead Licensing Engineer	T Chapman		
Approver	VP, Xe-100 Program Manager	MP van Staden		

CONTENTS

FIGURES

TABLES

ABBREVIATIONS

This list contains the abbreviations used in this document.

Abbreviation or Acronym	Definition
ACRS	Advisory Committee on Reactor Safeguards
ADUN	acid deficient uranyl nitrate
AGR	Advanced Gas Reactor
ANS	American Nuclear Society
AOO	Anticipated Operational Occurrence
ANSI	American National Standards Institute
ARDC	Advanced Reactor Design Criteria
ASME	American Society of Mechanical Engineers
ATR	Advanced Test Reactor [INL]
AVR	Arbeitsgemeinschaft Versuchsreaktor
BAF	Bacon anisotropy factor
BDBE	Beyond Design Basis Event
BISO	Bistructural Isotropic
CFR	Code of Federal Regulations
COL	Combined License
CVD	chemical vapor deposition
DBA	Design Basis Accident
DBE	Design Basis Event
DDN	Design Data Need
DLOFC	Depressurized Loss of Forced Cooling
DOE	Department of Energy
DTF	Designed-To-Fail
EAB	Exclusion Area Boundary
ECCS	Emergency Core Cooling System
EFPD	effective full-power days
EPA	Environmental Protection Agency
EPRI	Electric Power Research Institute
EPZ	Emergency Planning Zone
FFF	fuel fabrication facility

Abbreviation or Acronym	Definition
FIMA	Fissions per Initial Metal Atom
FOAK	first-of-a-kind
FQ	Fuel Qualification
FSV	Fort St. Vrain
GA	General Atomics
GASSAR	General Atomic Standard Safety Analysis Report
GDC	General Design Criteria
GT-MHR	Gas-Turbine Modular Helium Reactor
GWd/MTU	Gigawatt Days per Metric Tonne Uranium
GUI	Graphical User Interface
HCl	Hydrogen chloride
HFR	High-Flux Reactor
HM	Heavy Metal
HMTA	hexamethyltetramine
HPB	Helium Pressure Boundary
HRB	Hochtemperatur Reaktorbau
HTGR	High Temperature Gas-Cooled Reactor
HTR	High Temperature Reactor [German pebble-bed]
HTTR	High Temperature Engineering Test Reactor (Japan)
IAEA	International Atomic Energy Agency
INET	Institute of Nuclear and New Energy Technology, Tsinghua University - PRC
INL	Idaho National Laboratory
IPyC	Inner Pyrocarbon
KM	kernel migration
LBE	Licensing Basis Event
LEU	Low-Enriched Uranium (<20% U^{235} enriched)
LWR	Light Water Reactor
LPZ	Low Population Zone
MHTGR	Modular HTGR (General Atomics prismatic design)
MST	Mechanistic Source Term
MTR	materials test reactor

Abbreviation or Acronym	Definition
MTS	methyltrichlorosilane
MWe	Megawatt electric
MWt	Megawatt thermal
NGNP	Next Generation Nuclear Plant
NPR	New Production Reactor
NQA-1	Nuclear Quality Assurance
NRC	Nuclear Regulatory Commission
OPyC	Outer Pyrocarbon
ORNL	Oak Ridge National Laboratory
PAG	Protective Action Guide
PBMR	Pebble Bed Modular Reactor
PIE	post-irradiation examination
PIH	post-irradiation heating
PIRT	Phenomena Identification and Ranking Table
PRA	Probabilistic Risk Assessment
PSID	Preliminary Safety Information Document
R/B	release rate-to-birth rate ratio
RAI	Request for Additional Information
RCCS	Reactor Cavity Cooling System
RG	Regulatory Guide
RN	radionuclide
RPV	Reactor Pressure Vessel
SAFDL	specified acceptable fuel design limit
SAR	Safety Analysis Report
SARRDL	specified acceptable system radionuclide release design limit
SC-MHR	Steam Cycle Modular Helium Reactor
SiC	silicon carbide
SRM	Staff Requirements Memorandum
SRP	Standard Review Plan
TCE	trichloroethylene
TEDE	Total Effective Dose Equivalent

Abbreviation or Acronym	Definition
THTR	Thorium High Temperature Reactor
TRISO	Tri-structural Isotropic
UCO	uranium oxycarbide (admixture of UC_2 and UO_2)
V&V	verification and validation
VHTR	Very High Temperature Reactor
X-energy	X Energy, LLC

1. INTRODUCTION

X-energy is developing the helium-cooled Xe-100 modular High Temperature Gas-Cooled Reactor (HTGR) as an advanced nuclear power source [Braudt 2021] [1]. The Xe-100 modular HTGR utilizes a pebble-bed reactor core of the type that has been under development and operated internationally for five decades. The Xe-100 produces electrical power and/or process steam to serve the electricity generation market as well as non-traditional applications for nuclear energy, including secure government installations, small and/or isolated grids, and industrial process steam users.

Building upon the extensive investigations of Tri-structural ISOtropic (TRISO)-coated fuel particle performance beginning in the 1970s and the substantial progress made by the more recent U.S. Department of Energy (DOE) Advanced Gas Reactor (AGR) Fuel Development and Qualification Program, the Xe-100 Project has adopted the low-enriched (LEU) uranium oxycarbide (UCO)[1] TRISO-coated particle being qualified by the AGR program as the reference design. This design selection takes advantage of the demonstrated superior high temperature performance and high burnup capabilities of the LEU UCO particle compared to the LEU UO_2 particle that was used in the design of earlier pebble-bed modular HTGRs (Section 5.3.2).

The AGR program is focused upon qualifying UCO TRISO-coated particles in fuel compacts that are characteristic of prismatic fuel elements. The process conditions for making fuel spheres, also referred to as fuel pebbles, are different from those for making fuel compacts. In particular, the pressures for pressing spheres are higher than the pressures for molding fuel compacts (typically, 300 MPa vs. less than 20 MPa). This difference is, however, somewhat offset by the lower packing fraction of the particles in the sphere volume (~10%) compared with that of particles in the cylindrical compact volume (~35-45%). The lower packing fraction results in less likelihood of contact between the outer surfaces of particles. Consequently, X-energy will demonstrate that these fabrication process differences will not have any deleterious effects on the UCO TRISO-coated particles (a low probability concern since both HEU UCO and LEU UO_2 TRISO-coated particles in spheres have performed well [EPRI-AR-1] [1]). In addition, new equipment will be constructed for manufacturing fuel spheres in the U.S. [[

]]^{P} This report describes the approach that X-energy has adopted to incorporate and leverage the experience and results of the AGR Program and to demonstrate that UCO TRISO-coated fuel particles in spheres will meet the fuel quality and performance requirements derived from the top-level radionuclide[2] (RN) control requirements adopted by the Xe-100 design (Section 4).

The TRISO fuel particle consists of a microsphere (i.e., “kernel”) of nuclear material encapsulated by multiple layers of pyrocarbon and a silicon carbide layer. This multiple-coating-layer system has been engineered to retain the fission products (FP) generated by fission of the nuclear material in the kernel under normal operation and accident conditions. Although plant safety depends on many factors, the TRISO-coated fuel is particularly critical to the safe operation of the reactor because the fuel particles are

[1] “Uranium oxycarbide” is a shorthand term to denote an admixture of uranium dioxide and uranium carbide, the two phases prominently present in a UCO kernel.
[2] The terms “radionuclide” and “fission product” are used interchangeably throughout this paper although “radionuclide” is the more general term since it includes important activation products, such as Ag-110m, H-3, etc.

the primary (but not the only) barrier to fission-product release in modular HTGRs and provides the functional containment for the Xe-100 design[3].

Thus, the fuel's ability to retain fission products at the source in the core is extremely important to the safety case and licensing approach for modular HTGRs. It is a key part of developing mechanistic source terms [Loza 2021] [3] because accident analyses are predicated upon the high-quality fuel performing predictably during normal reactor operation and a broad spectrum of off-normal events that will be identified by Probabilistic Risk Assessment (PRA). Qualification of the fuel to stringent fuel performance requirements is essential for licensing these designs (Section 4).

1.1. PURPOSE

This report describes the fuel qualification approach for the Xe-100 design. Analogous to [NGNP FQ WP 2010] [4], the purpose of this report is as follows:

- Identify existing regulations, regulatory guidance, and licensing precedents relevant to the qualification of fuel for the Xe-100 design;
- Summarize existing understanding, data, and analysis methods regarding TRISO-coated particle fuel performance;
- Review reactor and fuel design and performance requirements;
- Describe the national and international experience base that provides the technical foundation for the fuel qualification approach;
- Describe planned fuel fabrication, irradiation, safety testing activities, and approach to qualify the reference fuel (UCO TRISO-coated particles); and
- Provide a documented basis for engagement with regulators on the planned approach and information required for qualifying the Xe-100 reference fuel.

Specific NRC review objectives are described in Section 7.

1.2. SCOPE

This report focuses on the existing U.S. licensing frameworks, recent licensing guidance related to advanced reactor fuel qualification, use of the existing fuel performance data, and the planned fuel qualification program to establish the bases of fuel performance under the intended operating conditions. The report will emphasize how the planned X-energy fuel qualification approach builds upon the success of the AGR program to develop and qualify UCO TRISO-coated particles.

[3] From SECY-18-0096: The term "functional containment" is applicable to advanced non-LWRs without a pressure retaining containment structure. A functional containment can be defined as "a barrier, or set of barriers taken together, that effectively limit the physical transport and release of radionuclides to the environment across a full range of normal operating conditions, AOOs [anticipated operational occurrences], and accident conditions."

1.3. INTERFACING REFERENCES

This licensing topical report on fuel qualification is one of several reports covering key regulatory issues that will be discussed with the NRC review as part of the Xe-100 pre-application engagement strategy. X-energy sees fuel qualification and the development of mechanistic source terms (MST) as tightly-coupled elements of the safety approach for the Xe-100 that directly contribute to implementation of the risk-informed, performance-based licensing basis approach being followed and as described in the guidance found in [NEI 18-04] [5] as endorsed and clarified by NRC Regulatory Guide (RG) 1.233 [NRC 2020] [6]. The issues that are most closely related to this report include:

- EPRI-led topical report on TRISO-coated particle fuel from the AGR-1/2 irradiation [EPRI-AR-1] [1] campaigns and the subsequent NRC staff Safety Evaluation (SE) on that report [EPRI-AR-1 FSER] [7];
- Mechanistic Source Term and Functional Containment Approach described in X-energy licensing topical report XE00-R-R1ZZ-RDZZ-L-000632 [Loza 2021] [3] of the same name;
- Probabilistic Risk Assessment (PRA) efforts being conducted using the trial-use standard (now in publication) ASME/ANS RA-S-1.4-2020 Probabilistic Risk Assessment Standard for Advanced Non-Light Water Reactor Nuclear Power Plants [RA-S-1.4-2021] [8]; and
- Licensing Basis Event (LBE) identification and selection, Structures, Systems, and Component (SSC) safety classification, and the evaluation of defense-in-depth adequacy methodology described in NEI 18-04 as implemented by X-energy, described in licensing topical report XE00-R-R1ZZ-RDZZ-L-000687 Risk-Informed Performance-Based Licensing Basis Approach for the Xe-100 Reactor [Vaughn 2021] [9].

Reviewers are also advised to reference the “Xe-100 Technology Description Technical Report” [Braudt 2021] [1] for details on the Xe-100 design and unit and plant operations. NRC staff review of that report is not requested at this time.

1.4. DOCUMENT LAYOUT

Section 2 provides background information on the Xe-100 reactor and the reference fuel design. Section 3 provides an overview of the regulatory bases that apply to TRISO fuel qualification for modular HTGR design and licensing. Section 4 describes the fuel design basis and the fuel requirements for the Xe-100 design. Section 5 summarizes the background information on the TRISO fuel technology base resulting from decades of international TRISO fuel technology development in the U.S., the United Kingdom, Germany, Russia, Japan, China, and South Africa. Section 6 summarizes the planned fuel qualification program supporting the Xe-100 design. Section 7 provides the NRC review outcome objectives.

There are also three appendices. Appendix A includes a tabulation of the “items for follow up” that were identified as part of the NRC staff’s review of the NGNP Fuel Qualification and Mechanistic Source Term white papers. Appendix B provides a description of the statistical Quality Control (QC) protocols used to assure that as-manufactured TRISO fuel has requisite properties defined in the fuel product and fuel process specifications. Appendix C describes [[

]]^{P}. It is anticipated that a future revision of this licensing topical report will include the remaining test and experimental data as they are developed to

affirm the conclusions made herein remain valid for TRISO-coated particle fuel manufactured and qualified using the methods and processes described.

1.5. OUTCOME OBJECTIVES

The information in this report serves as a basis for interaction with the NRC staff (and other regulatory agencies) on the topic of fuel qualification. This report focuses on the use of the AGR program UCO data base along with the national and international TRISO fuel-performance data base [[ ]]^{P} to establish the basis of fuel performance under the intended design conditions. The primary issues for which NRC staff review and approval are requested include:

- The fuel design and performance requirements in Section 4 are adequate for establishing an acceptable design basis to support the licensing of the Xe-100 reactor.
- Plans established in Section 6 for qualification of the UCO TRISO-coated particles in spheres are generally acceptable. These include:
  - Utilization of the AGR UCO compact data for normal operation and transient/accident heat-up conditions;
  - [[ ]]^{P}
  - [[ ]]^{P}
  - [[ ]]^{P}

2. BACKGROUND

2.1. XE-100 REACTOR DESIGN

The general HTGR concept evolved from early air-cooled and carbon dioxide (CO_2)-cooled reactors. The use of helium instead of air or CO_2 as the heat transport fluid in combination with ceramic fuel and a graphite moderator offered enhanced neutronic and thermal efficiencies and several advanced safety characteristics. The combination of helium and a ceramic core makes it possible to produce high temperature nuclear heat while maintaining a large safety margin to material limits. Two reactor core configurations, a pebble-bed core and a prismatic core, have been developed internationally for the commercial modular HTGR designs.

The Xe-100 reactor design is based on a pebble-bed core configuration. Pebble bed reactor technology dated back to the late 1960s, when the 46 MWt Arbeitsgemeinschaft Versuchsreaktor (AVR) was designed and operated in Germany. Later, advanced pebble bed reactor designs were developed in Germany, South Africa, and China. The Chinese have a modular HTGR pebble-bed reactor design, the HTR-PM, with two 250 MWt reactor modules serving a single 200 MWe turbine/generator [Zhou 2013] [10], that has completed cold commissioning as of October 2020 [WNN_HTR-PM] [11]. The Xe-100 reactor technology basis, design parameters, fueling scheme, pebble fuel, fuel handling and storage system, and safety characteristics are discussed in X-energy's core design reports [Mulder 2016] [12] and [Mulder 2021] [13].

The Xe-100 reactor and steam generator systems are shown in Figure 1. The main reactor characteristics, including dimensions, thermal power, and major operating conditions are given in Table 1. The active core volume is filled with approximately 224,000 spherical fuel elements, or pebbles, to form the pebble bed. The pebbles each contain approximately 19,000 TRISO-coated particles. The fuel particles consist of a fissionable ceramic fuel kernel surrounded by three ceramic coating layers for retention of fission products. Fissions within the coated particles create the nuclear heat which is conducted to the pebble's surface. A helium circulator transports the helium heat transport fluid through the pebble bed, transporting heat from the pebbles to the steam generator.

Fuel pebbles are loaded into the core while the reactor is operating through a central tube at the top of the reactor pressure vessel (RPV). A fuel discharge system at the bottom of the core removes spent pebbles through the bottom of the RPV for assessment by the Burn-up Measurement Systems (BUMS) that evaluates each pebble for physical damage and burn-up, and can return them to the top of the RPV, where they are again loaded into the core, or send them to spent fuel storage. A typical pebble goes through this process [[ ]]^{P} times before it is removed from the reactor before reaching burn-up limits. The spent fuel pebbles are inventoried and placed into spent fuel casks for storage.

The maximum fuel temperature during normal operation is limited to well below 1000°C, which is significantly lower than that of several earlier HTGR designs. The core excess reactivity is limited by on-line refueling, since fuel can be loaded and unloaded as desired during full power operation. The Xe-100 has an overall negative temperature coefficient of reactivity due to the Doppler broadening of the uranium kernel content, fuel pebble graphite, and reflector graphite, ensuring stability of operations and negative reactivity insertion during core heat-up events. This inherent reactivity feedback is one of the primary safety features the fuel is credited with during transient and safety analyses and allows the Xe-100 to achieve a safe shutdown condition for certain LBEs [Mulder 2021] [13].

Safe shutdown capability is also provided by two banks of control rods inserted into the side reflector. One control rod bank, the Reactivity Control System, is used in normal operation and can achieve hot shutdown if inserted. The second control rod bank, the Reactivity Shutdown System, is credited as the diverse reactivity control function. It is inserted by the safety-related Reactor Protection System and is used to establish long-term cold shutdown conditions. The relatively small core diameter allows safe shutdown by inserting control rods into the side reflectors only; no in-core control rods are needed [Mulder 2021] [13].

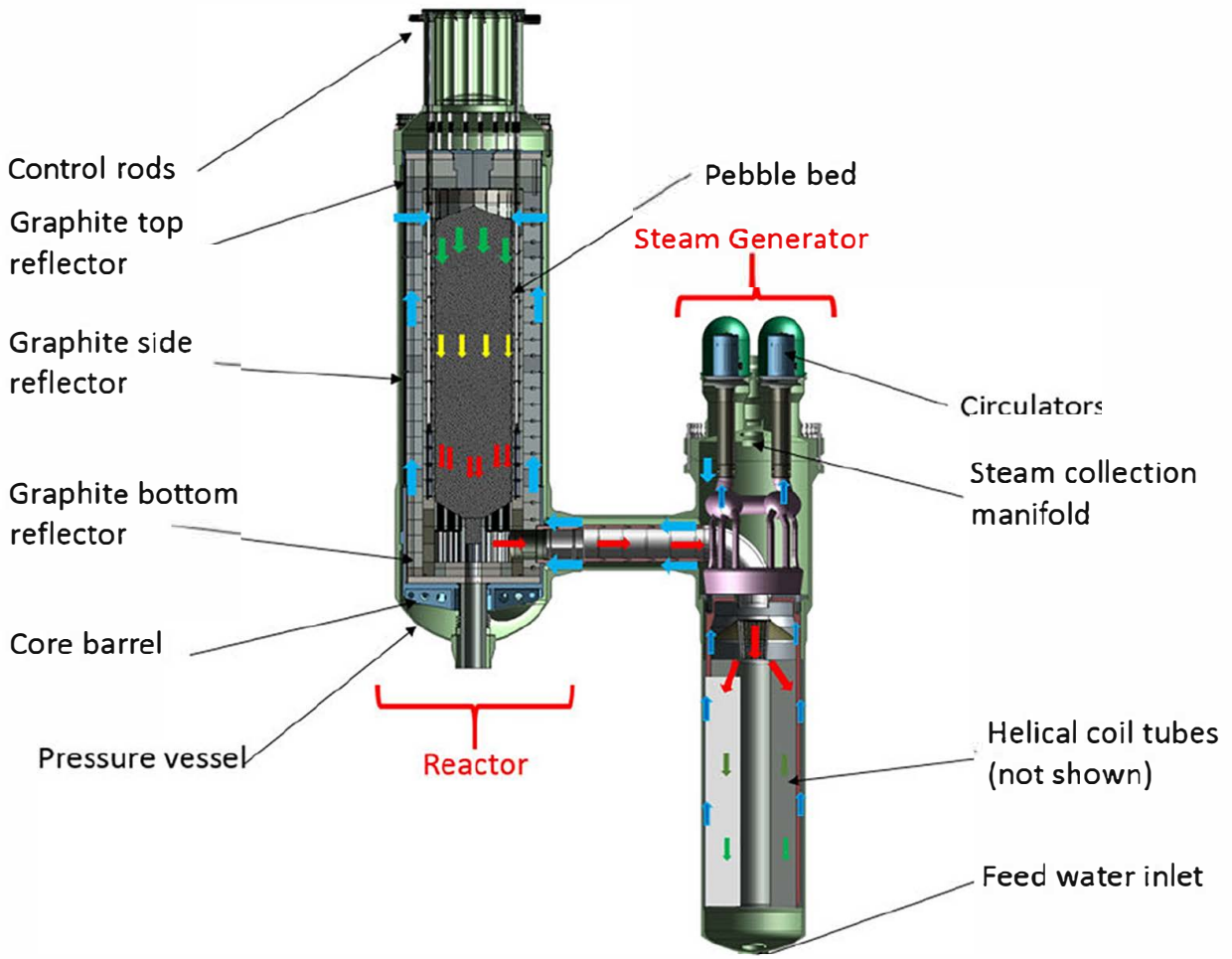

Figure 1: Xe100 Section View

Table 1. Xe-100 Reactor Main Characteristics

Thermal power	200 MWt	Helium flow rate (excluding bypass)	[[ ]]^{P}
Core volume	[[ ]]^{P}	Helium pressure	7 MPa
Core average power density	[[ ]]^{P}	Pebble packing fraction	[[ ]]^{P}

Height (flattened pebble bed)	[[ ]]^{P}	Particle packing fraction	[[ ]]^{P}
Diameter of pebble bed	[[ ]]^{P}	Burnup	168 GWd/MTU
Average pebble passes through the core	6	Enrichment	15.5%
Heavy metal loading per pebble	[[ ]]^{P}	Moderation ratio (C/U)	[[ ]]^{P}
TRISO-coated particles per pebble	~19,000	U-235 per pebble	[[ ]]^{P}
Gas inlet temperature	[[ ]]^{P}	Daily pebble charge	[[ ]]^{P}
Gas outlet temperature	750°C	Average fuel residence time	[[ ]]^{P}

If active cooling is lost, maximum fuel temperatures are limited by design, and decay heat is naturally removed from the fuel through the core structures, core barrel, and RPV via conduction, natural convection, and radiation. Heat removed from the RPV to the reactor cavity is discharged to the ultimate heat sink by one of three methods: 1) to the atmosphere by two trains of tube curtains called the Reactor Cavity Cooling System (RCCS) in an active-cooling mode, 2) to the atmosphere by the RCCS in a passive boil-off mode, or 3) directly to ground by conduction, natural convection, and radiation from the reactor cavity through the RCCS tube curtain and into (and through) the reactor building concrete structures. [[

]]^{P} Thermal transients in the Xe100 typically occur in periods of hours or days, rather than the seconds or minutes characteristic of LWR thermal transients. Further, the functions of moderation and thermal transport are fully separated in the case of the Xe-100 *versus* that of typical LWRs. Therefore, losing the thermal transport medium for any reason is of no safety consequence in the event of the Xe-100 as the permanent structural internals assume the thermal transport function. The moderator (pebble graphite matrix) remains in-place.

2.2. TRISO-COATED PARTICLE FUEL

Coated particle fuel has been used in HTGRs since their inception in the early 1960s [e.g., NGNP FQ WP 2010] [4] TRISO-coated particle fuel was first introduced in the Dragon reactor [Simon 2002] [13], and Fort St. Vrain (FSV) was the first electricity producing HTGR with an all TRISO-coated particle core [McEachern 2001] [15]. TRISO-coated particle fuel has been the fuel of choice for all modular HTGR designs, beginning with the German pebble bed HTR-Modul [Reutler 1984] [16] and the Modular High Temperature Gas-Cooled Reactor program in the U.S. [PSID 1992] [17].

In the Xe-100 design, the TRISO fuel particles consist of a UCO fuel microsphere ("kernel"), 425 µm in diameter and coated with multiple layers of pyrocarbon and silicon carbide (SiC) as shown in Figure 2. UCO was chosen for the Xe-100 design to take advantage of the ongoing AGR Fuel Development and Qualification Program at Idaho National Laboratory and Oak Ridge National Laboratory [AGR TDP 2016]

[18]. The AGR Program is focused on qualifying UCO TRISO fuel and is being conducted under a quality assurance program that meets Nuclear Quality Assurance (NQA-1) requirements [ASME 2012] [19] and has been reviewed and found by the NRC staff to be acceptable for use during the technology development phase of the NGNP Project [NRC 2012b] [20]. As summarized in [Petti 2016] [21], results to date have been excellent (Section 5.3.2). The performance of TRISO-coated particles made to AGR-1/2 specifications (identical coating structure, but different kernel and particle size, enrichment, and burn-up) [EPRI-AR-1] [1] was provided to the NRC staff for formal review in 2019 and received a favorable SE in August 2020 [EPRI-AR-1 FSER] [7].

While the Xe-100 design has evaluated several fuel specifications that could be used safely in operation, the reference Xe-100 fuel particle chosen is the same as the 15.5%-enriched, 425-µm UCO TRISO-coated particle irradiated in the AGR-5/6/7 qualification/margin test [Maki 2015] [22]. The finished particle nominal diameter is 855 µm. The different coating layers, consisting of the buffer, inner pyrolytic carbon (IPyC), SiC, and outer pyrolytic carbon (OPyC) layers are referred to collectively as a TRISO coating. The coating system constitutes a miniature multi-shell pressure vessel that provides retention of the fission products that are generated by fissioning of the nuclear material in the kernel. A substantial fraction of the fission products is retained inside the kernel itself. The performance of these coatings directly supports the functional containment approach for evaluating the Xe-100 design in transient and safety analyses.

The four coating layers of a TRISO-coated particle have specialized purposes as shown in Figure 2, but in composite they constitute a high-integrity pressure vessel that retains of fission products. The functions of the low-density buffer layer are to provide a reservoir for fission gases released from the fuel kernel, to attenuate fission recoils, and to accommodate kernel swelling under irradiation. The main functions of the high-density IPyC coating are to provide a smooth regular substrate for the deposition of a high integrity SiC coating and to prevent chlorine (Cl_2) and hydrogen chloride (HCl) from reacting with the fuel kernel during the SiC deposition process; hence, a major benefit of the IPyC coating is realized during fuel fabrication. The IPyC coating, which is intimately bonded to the SiC coating, also helps to maintain the SiC coating in compression, as the former shrinks under irradiation, while the latter is dimensionally stable.

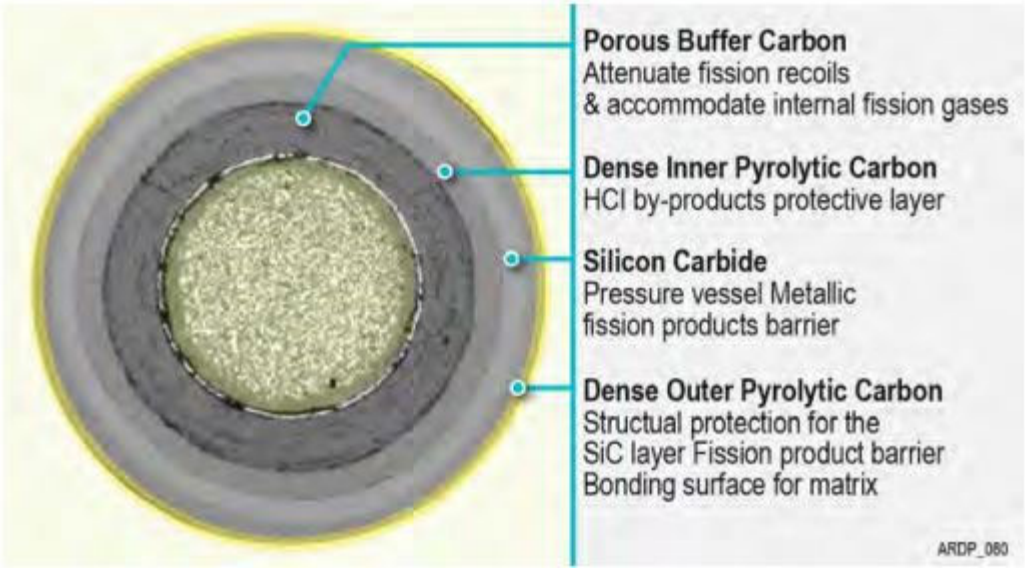

Figure 2: TRISO Fuel Particle Configuration and Coating Functions

The SiC coating is the most important in the TRISO coating system in terms of RN retention because it serves as the primary barrier to the release of fission products from the coated particle, particularly some of the volatile metallic fission products such as cesium. The high-density OPyC coating, which shrinks under irradiation, also generates a compressive stress in the dimensionally stable SiC, partially compensating for the tensile stress component induced by the internal gas pressure. The PyC coatings also effectively retain fission gases, including radiologically-dominant iodine isotopes, at temperatures up to about 1800°C for tens to hundreds of hours in fuel particles with defective (as-manufactured) or failed (in-service) SiC layers.

The physical properties and performance capabilities of the TRISO-coated particles are among the most important factors impacting the radiological safety of the HTGR. This is because fission product retention in the fuel, as well as the high fuel burnups and temperatures that can be tolerated in the reactor core, are primarily determined by the fuel particle properties [NGNP FQ WP 2010] [4].

2.3. PEBBLE FUEL DESIGN FOR THE XE-100

The defining characteristic of the pebble-bed reactor and the key to the safety and operational simplicity of the Xe-100 is the use of TRISO-coated fuel particles embedded in fuel spheres, or pebbles. The design of the coated particles and fuel sphere, including their nominal dimensions, is depicted in Figure 3.

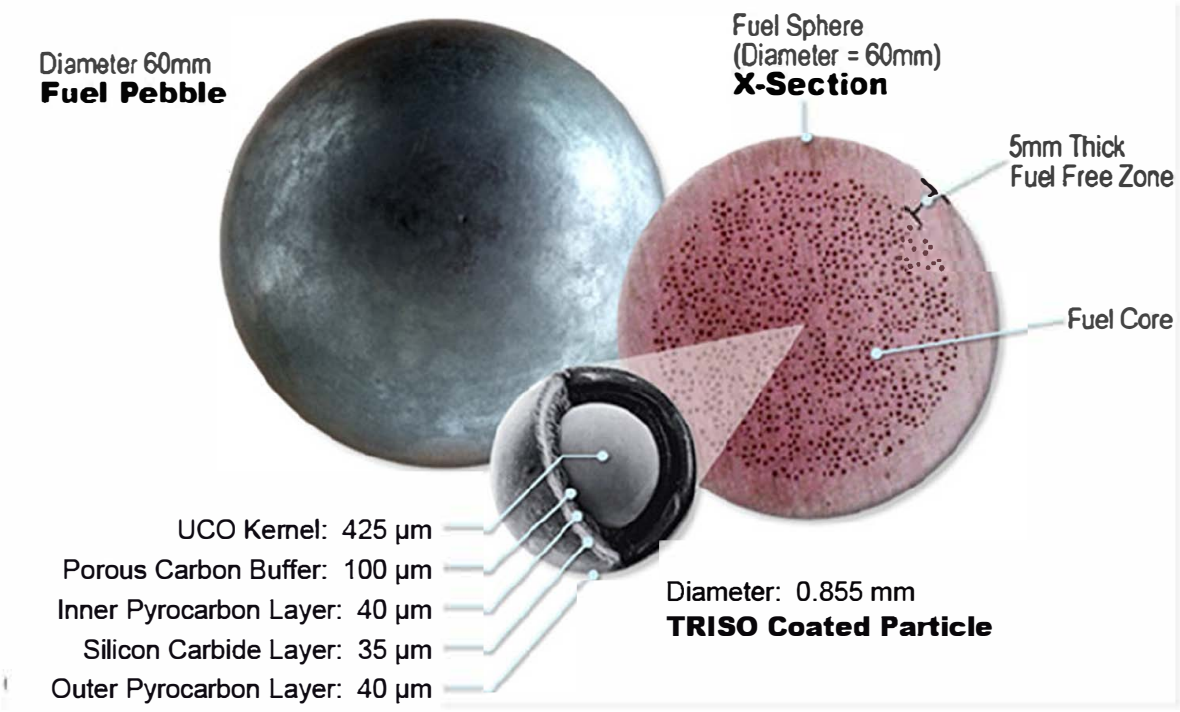

Figure 3: Xe100 Reference Fuel Element Design

As shown in Figure 3, the 60 mm-diameter spherical fuel element consists of two zones: the inner spherical region is known as the fuel zone, and the outer shell surrounding the fuel region known as the fuel free zone. The 50-mm diameter fuel zone of each fuel sphere contains thousands of TRISO-coated fuel particles evenly distributed in a carbonaceous matrix material. The fuel-free zone is a 5 mm thick shell of the same matrix material formed by a high-pressure isostatic pressing process and machined to final dimensions. X-energy has selected the U.S. equivalent of German A3-3 ("US A3-3") matrix as the reference matrix for the Xe-100 fuel element. The relatively low particle packing fraction in the fuel zone [[ ]]^{p} and pebbles being fully wetted by the helium everywhere, leads to lower temperature gradients across the fuel particle as compared to prismatic fuel designs and lower maximum fuel temperatures during normal operation.

Pebble fuel technology is well developed with a strong pedigree and operating experience from the international community [e.g., TEC-1674 2012] [23]. The spherical fuel system was initially developed, licensed, and manufactured during the German High Temperature Reactor (HTR) fuel development program. More than a million spherical fuel elements were produced and used to power the German AVR test reactor [VDI-Verlag 1990] [24] at the Research Center Jülich and the Thorium High-Temperature Reactor (THTR), a German demonstration reactor at Hamm-Uentrop [Baeumer 1991] [25]. The German fuel development program was later extended for the performance envelope of the HTR-MODUL (High-Temperature Reactor Module, a German concept designed by Siemens) that later became the basis for all the design evolutions of the Pebble Bed Modular Reactor (PBMR[4]) Project in South Africa and the technical basis for the HTR-PM, presently being commissioned in China. Real-time irradiation tests in the

[4] In this document, PBMR will refer solely to the South African company model. "Pebble bed reactor" with no abbreviation will be used for all other discussion

prototypical neutron spectrum environment of the AVR and accelerated materials reactor tests have confirmed the superior irradiation performance of pebble fuel.

2.4. THE XE-100 SAFETY DESIGN APPROACH

2.4.1. Objectives of the Safety Design Approach

Per the NEI 18-04 definition, the Xe-100 safety design approach comprises the design strategies that support safe operation of the plant and control the risks associated with unplanned releases of radioactive material and protection of the public and plant workers. These include the use of robust barriers, multiple layers of defense, redundancy, and diversity, and the use of inherent and passive design features to perform safety functions. These strategies apply throughout the spectrum of Licensing Basis Events (LBEs): normal operating conditions including Anticipated Operational Occurrences (AOOs), Design Basis Events (DBEs), Design Basis Accidents (DBAs) which are analyzed assuming only the safety-related SSCs are available, and Beyond Design Basis Events (BDBEs). The Xe-100 safety design approach is different from that of the currently licensed LWRs, which focuses on preventing and mitigating core damage and a large early release of radionuclides in the event of core damage. The safety design approach of the Xe-100 precludes fuel degradation or failure sufficient to significantly affect radiological consequences and focuses on preventing and limiting the release of relatively small amounts of radionuclides during normal operation and off-normal event sequences. X-energy uses the guidance of NEI 18-04, as clarified by Regulatory Guide 1.233, as the basis for identifying and selecting LBEs for evaluation in the design and licensing bases for the Xe-100. The implementation of NEI 18-04 is further described in X-energy's licensing topical report "Risk-Informed, Performance-Based Licensing Basis Approach for the Xe-100 Reactor" [XE00-R-R1ZZ-RDZZ-X-000687] [Vaughn 2021] [9].

NEI 97-04 was developed to help utilities organize and collate design bases information and supporting design information. NRC Regulatory Guide 1.186, "Guidance and Examples for Identifying 10 CFR 50.2 Design Bases," (Rev. 0, December 2000) endorses the NEI 97-04 revised Appendix B of the same title and provides guidance in generating the Xe-100 design bases, including the appropriate information related to fuel fabrication, qualification, and performance.

2.4.2. Functional Containment Approach for the Xe-100

A distinctive difference between the LWR and Xe-100 safety design approaches is the credited principal barriers to release of radionuclides to the environment and therefore, how defense-in-depth is demonstrated. In the current LWR designs, the principal barrier to RN release credited during severe accidents is the pressure-retaining, low-leakage (i.e., essentially leak-tight) containment building. The limiting LBE for the LWR is the loss-of-coolant accident resulting from a breach of the primary coolant system. This postulated accident sequence is a rapid transient of seconds to minutes in duration characterized by high energy release of high temperature, pressurized-water primary coolant into the containment, which is assumed to result in significant damage to the oxide fuel pellets and zirconium cladding, followed later by hydrogen generation from fuel clad/steam interaction and potential hydrogen detonation. Since the initiating event is a breech in the primary coolant circuit, it is assumed that all but

one of the principal barriers to RN release are compromised: the fuel cladding and reactor coolant pressure boundary. These characteristics require the containment building to absorb the stored energy of the coolant system, to absorb the energy of potential detonation of hydrogen, and to contain RNs released from the fuel, all reliant on the integrity of its design basis functions of pressure-retention and low-leak rates.

In contrast to LWRs, modular HTGRs use a functional containment approach as part of their design bases, comprised of five RN release barriers. These barriers are designed to limit RN release from the core to the environment to insignificant levels during normal operation and across a broad spectrum of postulated accidents. The various phenomena that determine RN transport and release for a spectrum of postulated accidents are shown schematically in Figure 4 [55], comprised of the five principal release barriers: (1) the fuel kernel; (2) the particle coatings, particularly the SiC coating; (3) the spherical fuel-element matrix, including the fuel free zone; (4) the reactor helium pressure boundary; and (5) the reactor building. Each of these barriers contributes to limiting the release of RNs to the environment to meet the top-level radiological criteria. The contribution of each of the barriers in limiting RN release to the environment is calculated for each postulated event, depending on the response of the reactor to the event. The contribution of each barrier is design specific and specific both to the event scenario and the radionuclide species. These phenomena are described in greater detail in [Loza 2021] [3] as part of X-energy's development of event-specific mechanistic source terms.

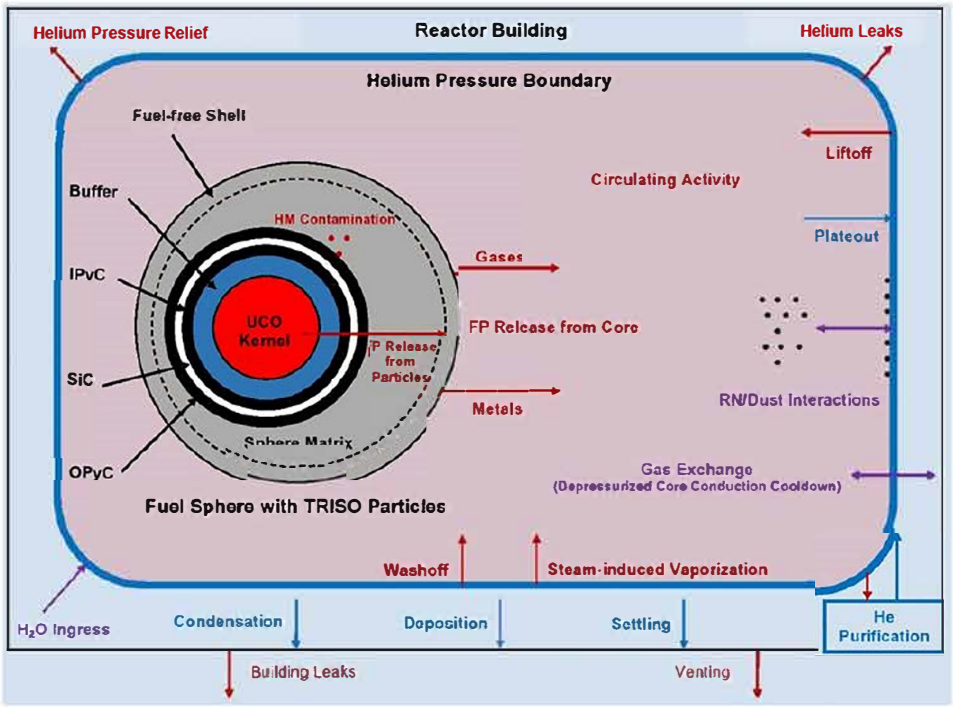

Figure 4: Pebble Bed HTGR Radionuclide Retention System

Collectively, the spherical fuel element comprises the first three release barriers. Typically, the fuel particles retain >99.999% of the fission products even during BDBEs. Therefore, the fuel is actually the "containment." A major difference between the Xe-100 and typical LWRs is that failure of the RPV, a pebble being broken, or TRISO-coated particles failing does not have an impact on any of the neighboring radionuclide barriers, whereas in LWRs these boundaries are all connected.

The release fraction from the particles is calculated accounting for the as-manufactured fuel quality (e.g., the allowable heavy-metal contamination and coating defects), fuel failure during irradiation, incremental fuel failure under accident conditions, and diffusion of fission products through the intact particle coatings under normal operating and accident conditions. These factors are calculated by the X-energy code suite XSTERM on an event specific basis, depending on the burnup of the fuel, maximum operating temperature, and maximum temperature reached in the accident and, where applicable, air and/or water contamination in the reactor coolant, (e.g., as a result of a steam generator tube rupture event sequence). The spherical fuel matrix provides retention of metallic fission products (e.g., retention of cesium by a factor of 10 and strontium by a factor of more than 100). The matrix is essentially non-retentive for the gaseous fission products (e.g., iodine and noble gases). The performance of the ex-core RN release barriers (the reactor helium pressure boundary and the reactor building) is described in [Loza 2021] [3].

2.4.3. Reactor Passive and Inherent Design Characteristics that Contribute to the Design Bases

Since their inception [e.g., Reutler 1984] [16], modular HTGRs have been characterized by a strong emphasis on passive safety, based upon the inherent characteristics of the ceramic reactor core embedded in a steel RPV and the TRISO-coated particle fuel. The hallmark feature of modular HTGRs is to design the reactor and fuel such that the radionuclides are largely retained in the fuel during normal operation and a broad spectrum of off-normal events. The inherent characteristics of the modular HTGR reactor core and coated-particle fuel facilitate a design that has unparalleled safety features compared to other advanced reactor designs.

Passive design features are defined as design features engineered to meet their required functional design criteria without (a) needing successful operation of active systems with mechanical components such as pumps; blowers; heating, ventilation, and air conditioning or sprays that require an external power source; (b) depending on alternating current electric power; or (c) relying on operator actions. Inherent design characteristics are those characteristics associated with the reactor concept and the properties of the materials selected for the basic reactor components [NGNP DID WP 2009] [27]. Of direct relevance here are those passive design features and inherent characteristics that serve to limit the fuel temperatures during normal operation and off-normal events such that the fuel integrity is not compromised (Section 2.1).

In addition to the fuel, the specific characteristics of the Xe-100 design that contribute to safety include:

- A large solid graphite moderator/reflector structure with very high temperature capability. The graphite provides large heat capacity in the core that increases the time constants and reduces the magnitude of core thermal transients. Limiting transients occur over hours and days, not seconds. No fast-acting active safety systems are required to maintain the fuel within specified acceptable fuel design limits.
- A passive heat transfer path from the fuel to the ultimate heat sink, the external atmosphere or ground. This heat transfer path through graphite moderator/reflector and through the reactor vessel to the reactor cavity cooling system, and to the external atmosphere has the capacity, without requiring any active systems, to limit fuel, reactor pressure vessel, and reactor cavity structural

concrete temperatures so that degradation of the fuel-element barriers is limited to acceptable levels and there is no degradation of the core geometry.

- A large negative temperature coefficient that inherently limits reactor power levels to relatively low levels under accident conditions without control rod or reserve shutdown system insertion of negative reactivity.
- A low core power density and high core surface-to-volume ratio that limits the fuel temperature rise during the most limiting conditions of loss-of-forced cooling and depressurization of the primary circuit.
- A single-phase, chemically inert, neutronically transparent, and high thermal conductivity helium heat transport fluid with low stored energy, minimizing the functional requirement for containment of energy in a postulated breach of the reactor helium pressure boundary.

3. OVERVIEW OF REGULATORY REQUIREMENTS AND GUIDANCE

This section identifies currently applicable NRC regulations and guidance, relative to fuel qualification, that must be satisfied for licensing the Xe-100 in the U.S. and cites related precedents. In applying NRC regulations, which evolved primarily to address LWR safety design approaches, to modular HTGRs it is necessary, in some cases, to consider the basis for the regulations and anticipate necessary exemptions due to the basic differences between LWR and HTGR technology. Recently, NRC, DOE, and the industry have proactively addressed regulatory gaps associated with advanced reactors as well as gaps of a generic nature stemming from LWR-based small modular reactors that are relevant to advanced non-light water reactors as well. A review of applicable regulations, policies, and guidance related to fuel performance, specification, and qualification is provided here. The applicability of regulations, policies, and guidance to the Xe-100 design collectively will be assessed in specific licensing applications of Title 10 of the Code of Federal Regulations (10 CFR) with supported exemptions requests as necessary.

3.1. NRC REGULATIONS

Current NRC regulations contain little information that is directly pertinent to the general process of - or specific requirements for - fuel qualification. Instead, the regulations generally focus on requirements for in-reactor fuel performance.

Regulations pertinent to reactor fuel performance during normal operation, including anticipated operational occurrences (AOOs), are codified primarily in the General Design Criteria (GDC) contained in Appendix A to 10 CFR Part 50. The GDC for water-cooled nuclear power plants establish minimum requirements for the principal design criteria required to be developed for licensing applications. The GDC, which were developed in the context of LWR designs, may not directly apply to other types of reactors, such as modular HTGRs. Nevertheless, as discussed in the introduction to Appendix A of 10 CFR Part 50, the GDC are considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the principal design criteria for such units.

Relative to reactor fuel design, GDC 10, *Reactor Design*, states that:

> *…the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.*

Other criteria in Appendix A provide requirements for operation of systems to ensure that specified acceptable fuel design limits (SAFDLs) of GDC 10 are not exceeded during normal operation and AOOs (GDCs 12, 17, 20, 25, 26, 33, and 34). GDC 35 provides requirements for performance of systems, such as the LWR emergency core cooling system (ECCS), to ensure that fuel damage that could interfere with effective core cooling under accident conditions is prevented.

Recognizing the challenge of interpreting the GDCs, developed over decades of experience for LWRs, the NRC and the DOE engaged in a cooperative program to develop elements for an advanced reactor licensing framework. One of the first focus areas of this program was the development of Advanced Reactor Design Criteria (ARDC) for application to advanced non-LWRs. The effort included development of design criteria that are specific to modular HTGR designs.

DOE submitted a proposal for the content of the ARDCs to the NRC [INL 2014b] [28] in 2014, and the NRC staff published its initial comments on those criteria [NRC 2015] [29] in 2015. A draft Regulatory Guide on design criteria for non-LWRs was released by the NRC for public review and comment [NRC 2017] [30] in 2017. The result of public feedback and further review was the publication of Regulatory Guide 1.232 "Guidance for Developing Principle Design Criteria for Non-Light-Water Reactors" issued in April 2018. These advanced reactor design criteria include an alternative fuel performance design limit that is appropriate for modular HTGRs.

The quality assurance requirements in Appendix B of 10 CFR Part 50 apply to tests conducted to qualify fuel, to fuel fabrication, and to fuel performance analyses. This regulation is directly applicable to the Xe-100 and also captured as mHTGR-Design Criterion 1 of the ARDC.

mHTGR-DC 10 was adapted from GDC 10 as follows:

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

The following portions of the rationale for adaptation were provided:

> *Design features within the reactor system must ensure that the SARRDLS are not exceeded during normal operations and AOOs. The (TRISO) fuel used in the MHTGR design is the primary fission product barrier and is expected to have a very low incremental fission product release during AOOs.*
>
> *As noted in NUREG-1338 and in the NRC staff's feedback on the Next Generation Nuclear Plant (NGNP) project white paper, "Next-Generation Nuclear Plant – Assessment of Key Licensing Issues" the TRISO fuel fission product transport and retention behavior under all expected operating conditions is the key to meeting dose limits, as a different approach to defense in depth is employed in an MHTGR. The SARRDL concept allows for some small increase in circulating radionuclide inventory during an AOO. To ensure the SARRDL is not violated during an AOO, a normal operation radionuclide inventory limit must also be established (i.e., appropriate margin). The radionuclide activity circulating within the helium coolant boundary is continuously monitored such that the normal operation limits and SARRDLs are not exceeded.*
>
> *The SARRDLs will be established so that the most limiting license-basis event does not exceed the siting regulatory dose limits criteria at the exclusion area boundary (EAB) and low-population zone (LPZ), and also so that the 10 CFR 20.1301 annualized dose limits to the public are not exceeded at the EAB for normal operation and AOOs.*

mHTGR-DC 16 provides guidance for the PDC related to functional containment, which is one of the primary functions of the TRISO coated particles:

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

The following portions of the rationale for adaptation were also provided:

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

Functional containment is relied upon to ensure that dose at the site boundary as a consequence of postulated accidents meets regulatory limits. Traditional containment structures also provide the reactor and SSCs important to safety inside the containment structure protection against accidents related to external hazards (e.g., turbine missiles, flooding, aircraft). The MHTGR functional containment safety design objective is to meet 10 CFR 50.34, 52.79, 52.137, or 52.157 offsite dose requirements at the plant's exclusion area boundary (EAB) with margins.

The NRC staff has brought the issue of functional containment to the Commission, and the Commission has found it generally acceptable, as indicated in the staff requirements memoranda (SRM) to SECY-93-092 and SECY-03-0047. In the SRM to SECY-03-0047 (Ref. 10), the Commission instructed the staff to "…develop performance requirements and criteria working closely with industry experts (e.g., designers, EPRI, etc.) and other stakeholders regarding options in this area, taking into account such features as core, fuel, and cooling systems design," and directed the staff to submit options and recommendations to the Commission for a policy decision.

The NRC staff also provided feedback to the DOE on this issue as part of the NGNP project. In the NRC staff's "Next Generation Nuclear Plant — Assessment of Key Licensing Issues" , the area on functional containment and fuel development and qualification noted that "…approval of the proposed approach to functional containment for the MHTGR concept, with its emphasis on passive safety features and radionuclide retention within the fuel over a broad spectrum of off-normal conditions, would necessitate that the required fuel particle performance capabilities be demonstrated with a high degree of certainty."

X-energy's approach to addressing the functional containment performance criteria is provided in the companion paper "Implementation of Functional Containment Report," XE00-N-G1ZZ-GLZZ-D-000862 [Wiebe 2021] [31] and the mechanistic source term topical report companion to this one. This paper describes the approach to establishing the fuel specifications and associated fuel qualification methodology for meeting those specifications. The objective of this paper was familiarization for the Canadian Nuclear Safety Commission staff with the functional containment concept. The decomposition of functions traditionally performed by the containment building but assigned to other SSCs in the Xe-100 will be described in specific licensing applications.

Other fuel performance requirements are provided in 10 CFR 50.46, acceptance criteria for emergency core cooling systems for light water reactors (LWRs). 10 CFR 50.46 requires light-water power reactors fueled with uranium oxide pellets within cylindrical zircaloy cladding to be provided with an emergency core cooling system (ECCS). The ECCS will be designed so that its calculated cooling performance following postulated loss-of-coolant accidents conforms to specified criteria regarding peak cladding temperature, maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long-term cooling. However, modular HTGRs such as the Xe-100 are designed not to need a system analogous to the LWR ECCS. Passive, inherent heat removal capability is provided due to the selection of fuel design

characteristics (i.e., low power density, high thermal conductivity materials, pebble size and geometry) that assure the fuel pebbles remain within their performance envelope across the range of LBEs.

10 CFR 50.34 and 10 CFR 52.79, technical contents of applications for operating licenses (50.34) and COLs (52.79) provide some guidance on the content of the license applications for designs that differ significantly from LWR designs licensed before 1997, or that utilize simplified, inherent, passive, or other innovative means to accomplish their safety functions. Each licensing pathway references 10 CFR 50.43(e) which, in summary, requires a combination of analyses and test programs to demonstrate the performance of safety features and assure that sufficient data exist to assess the analytical tools used for safety analyses. This regulation applies to the Xe-100 design and for the fuel system (i.e., TRISO-coated particle fuel in pebble form) is addressed by the approach described in this topical report.

3.2. NRC POLICY STATEMENTS

No NRC policy statements directly apply to the type of fuel proposed for use in the Xe-100 or address testing or monitoring of the fuel, nor does the NRC policy statement on regulation of advanced nuclear power plants [NRC 2008] [32] explicitly address nuclear fuel qualification. However, NRC policy issues specific to the modular HTGR are identified in SECY-93-092 [NRC 1993] [33] and in Section 5 of NUREG-1338 [NRC 1995] [38], and the methodology for functional containment performance criteria is further developed in SECY-18-0096 and associated SRM. Of the 10 policy issues identified in SECY-93-092, both the “Containment Performance” and “Source Term” policy issues are related to the fuel because the use of the HTGR’s multi-barrier functional containment approach and associated mechanistic source terms for accident analyses are based on the performance of the fuel being both excellent and predictable.

3.2.1. Functional Containment Performance Criteria

Fundamental to the modular HTGR is its emphasis on RN release prevention by utilizing high-integrity fuel particles in a multi-barrier attenuation approach rather than a single, essentially leak-tight containment barrier to minimize radionuclide releases to the environment. The current LWR containment leakage requirements are currently described in GDC 16 and Appendix J of 10 CFR Part 50. The Xe-100 design adopts an approach that provides functional containment performance criteria to be met by the TRISO-coated particle fuel, the fuel pebble, the helium pressure boundary, and the reactor building depending on the analyzed transient or safety analysis.

In the Staff Requirements Memo (SRM) to SECY-03-0047 [NRC 2003] [35], the Commission directed the NRC staff to develop performance standards for approaches to radionuclide retention other than an essentially leak-tight containment barrier. In SECY-04-0103 [NRC 2004] [36] and SECY-05-0006 [NRC 2005] [37], the staff updated the Commission on the development of a performance standard based on functional containment performance criteria to evaluate the acceptability of the proposed designs, rather than relying on prescriptive containment design criteria. As discussed in Section 3.5.6, following its review of NGNP white papers on fuel qualification and mechanistic source terms, the NRC staff found the NGNP proposed modular HTGR functional containment performance standard to be reasonable [NRC 2014] [43]. In SECY-18-0096 [NRC 2018] [44] the NRC staff provided its proposed methodology for an applicant to develop plant-level design and performance criteria for SSCs based on accepted event categories and demonstrating the design meets the fundamental safety functions as described in what is now the

guidance of NEI 18-04. This methodology provides for event-sequence-specific crediting of the barrier function of certain SSCs for some or all event categories to demonstrate the design is capable of maintaining adequate retention of RNs. The NRC approved this methodology in an associated notation vote and SRM.

SECY-18-0096 provides further guidance on implementing the functional containment approach anchored on the definitions of functional containment provided in mHTGR-DC 16 (quoted above). The methodology is, in many ways, a forerunner to the guidance issued in NEI 18-04 and endorsed in RG 1.233. The identification and categorization of LBEs for the Xe-100 is done in a systematic and comprehensive manner using an Advanced Non-Light-Water Reactor (ANLWR) PRA to establish event sequences, estimate frequencies of occurrence, and uses the mechanistic source term approach to develop potential consequences for each LBE. These consequences are compared to established acceptance criteria in the form of a frequency-consequence (F-C) curve. The NEI 18-04 approach also provides consideration of both probabilistic and deterministic methods to establish the spectrum of LBE consequences. In the Xe-100 integrated transient and safety analyses, provided in each licensing application, each LBE will be described to include which SSCs are credited with specific barrier functions, the performance criteria the SSCs is credited with achieving, and the acceptance criteria (i.e., SAFDLs or SARRDLs) the SSCs credited in the event sequence are demonstrated to meet. Each licensing application will also include an evaluation of the adequacy of defense-in-depth as described in NEI 18-04 and implemented by X-energy as described in the companion topical report [Vaughn 2021] [9].

The other required functions (safety or otherwise) performed by traditional containment buildings are decomposed to other SSCs in the Xe-100 and will be described in their respective chapter/section of the licensing application. Examples of these functions include:

- Structural support to the helium pressure boundary and reactor system components,
- Geometry assurance to maintain inherent heat removal capability,
- Protection against external hazards, including aircraft impacts,
- Physical security, and
- Severe accident mitigation design alternatives credited in environmental assessments.

3.2.2. Source Term

The radiological source term for HTGR designs is defined as the quantities, timing, physical and chemical forms, and thermal energy of RNs released from the reactor building to the environment during certain postulated LBEs. The HTGR definition is judged appropriate for greater emphasis on fuel retention of radionuclides for event sequences rather than reactor building retention following an event.

In SECY-03-0047 [35], 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 performance of the reactor and fuel under normal and off-normal conditions is sufficiently understood to permit a mechanistic analysis. The Commission stated that sufficient data should exist on fuel performance through research, development, and testing programs to provide adequate confidence in this approach. The use of a MST approach is further evaluated in SECY-16-0012 [NRC 2016] [46], the ANLWR PRA standard developed by the American Society of Mechanical Engineers / American Nuclear Society (ASME/ANS) for trial use [RA-

S-1.4-2021] [8], and [NEI 18-04] [5] as endorsed by [NRC 2020] [6]. This closely related topic is further discussed in X-energy's MST licensing topical report [Loza 2021] [3].

3.3. NRC GUIDANCE/REFERENCES

3.3.1. NUREG-0111, "Evaluation of High Temperature Gas-Cooled Reactor Particle Coating Failure Models and Data"

NUREG-0111 [Tokar 1976] [47] addresses fuel particle coating failure models and data for highly enriched uranium (HEU) UC_2 TRISO fissile particles with a 200-µm kernel and ThO_2 bi-structural isotropic (BISO) fertile particles with a 500-µm kernel for service in a large prismatic HTGR of the kind that was being developed by General Atomics in the mid-1970s. Major differences in particle design, fabrication specifications, service conditions, and fuel performance requirements relative to the fuel for the Xe-100 limit the applicability of this report to the current low enriched uranium (LEU) fuel. The content of NUREG-0111 has largely been overcome by events in coated particle fuel development over the last 40 years. Nonetheless, experience with this and other diverse fuel types over the course of HTGR fuel development has provided valuable insights into the development and understanding of the LEU UCO TRISO fuel to be used in the Xe-100 reactor.

3.3.2 NUREG-0800, Standard Review Plan (LWR Edition), Section 4.2, "Fuel System Design"

The purpose of the fuel system design safety review under NUREG-0800 Standard Review Plan (SRP) Section 4.2 [NRC 2007] [48] is to ensure that the fuel design meets the LWR requirements of GDC 10 and 35 and the core coolability requirements of 10 CFR 50.46. To this end, SRP 4.2 contains guidance on evaluating SAFDLs that ensure that (1) LWR fuel is not damaged as a result of normal operation and anticipated operational occurrences, (2) fuel damage is never so severe as to prevent control-rod insertion when it is required, (3) the number of fuel-rod failures is not underestimated for postulated accidents, and (4) coolability is always maintained. The SAFDLs addressed in SRP Section 4.2 are specific to LWRs using UO_2 ceramic fuel in zirconium cladding as the fuel system. As discussed in Section 3.1, an alternative approach to a fuel performance design limit for modular HTGR coated particle fuel has been developed and endorsed by the NRC in the form of the ARDC and for implementation in LBE evaluations as described in [NEI 18-04] [5].

To demonstrate that the SAFDLs have been established and satisfied, SRP 4.2 states that the NRC staff will review:

- Design basis for the fuel
- Description and design drawings for the fuel
- Evaluation of the fuel design
- Plans for fuel testing, inspection, and surveillance.

Although these review areas are very general, most of the technical details contained in SRP 4.2 relative to these four review areas are very specific to LWR fuel, focused on parameters such as stress, strain, or

loading limits for spacer grids, guide tubes, thimble, fuel rods, control rods, channel boxes, and other fuel system structural members that have no analogy in modular HTGR fuel.

SRP 4.2 acknowledges that some design bases and related parameters can only be evaluated analytically. However, the analytical models discussed in SRP 4.2 are very specific to LWR fuel. In NUREG-1338 (Pages 4-6 and 4-8), the NRC staff relied upon the NUREG-0111 evaluation of particle coating failure models to guide its evaluation. However, as discussed in Section 3.3, NUREG-0111 has largely been overcome by events in coated particle fuel development over the last 40 years.

SRP 4.2 states that for a fuel design that introduces new features, a more detailed surveillance program, commensurate with the nature of the changes, is warranted. The program should include appropriate qualitative and quantitative inspections to be carried out at interim and end-of-life refueling outages. This surveillance program should be coordinated with prototype testing. When prototype testing cannot be performed, a special detailed surveillance program should be planned for the first irradiation of a new design. For a pebble-bed design like the Xe-100, the online recirculation of spherical fuel elements that includes burn-up measurement and the ability to monitor the circulating helium for RNs provides a practical approach for surveying fuel performance. It is anticipated that a detailed monitoring program will be described in the equivalent “Initial Startup and Test Program” chapter/section of an Xe-100 licensing application to define specific objectives of such monitoring, durations, frequencies, and other necessary details for NRC review.

During the NGNP Project preapplication regulatory review, discussed in Section 3.5.6, Oak Ridge National Laboratory (ORNL), under contract to the NRC, began preparation of selected sections, including Section 4 for the fuel system, of a Design Specific Review Standard (DSRS) for the NGNP modular HTGR. The SRP was used as the starting point from which to develop the DSRS. The ORNL effort resulted in an early draft, but the effort ended when the NGNP Program ended. The ORNL draft was not released for review by stake holders but may provide additional considerations of appropriate fuel system performance monitoring.

3.3.3 TRISO-Coated Particle Fuel Phenomenon Identification and Ranking Tables

In anticipation of future licensing applications for HTGRs, in 2006 the NRC commissioned a panel to identify and rank the phenomena associated with TRISO-coated-particle fuel to obtain a better understanding of the significant features of TRISO-coated-particle fuel design, manufacture, and behavior during both normal reactor operation and accidents documented in NUREG/CR-6844, Vol. 1 [PIRT 2004] [49]. Six Phenomena Identification and Ranking Tables (PIRTs) were developed by the panel, including PIRTs on:

- Manufacturing,
- Operations,
- Depressurized heat-up accident,
- Reactivity accident,
- Depressurized accident with water ingress, and
- Depressurization accident with air ingress.

In preparing the PIRTs, the panel assumed the plant to be a pebble-bed reactor with UO_2 fuel, except for the reactivity accident PIRT, in which case a prismatic reactor was considered. However, the panel also identified and evaluated the importance and knowledge rankings that would be different for prismatic reactor UCO fuel.

According to NUREG/CR-6844, the NRC would use the PIRT results to:

1. Identify key attributes of gas-cooled reactor fuel manufacture that may require regulatory oversight,
2. Provide a valuable reference for the review of vendor HTGR fuel qualification plans,
3. Provide insights for developing plans for fuel safety margin testing,
4. Assist in defining test data needs for the development of fuel performance and fission-product transport models,
5. Inform decisions regarding the development of NRC's independent HTGR fuel performance code and fission-product transport models,
6. Support the development of NRC's independent models for source term calculations, and
7. Provide insights for the review of vendor HTGR fuel safety analyses.

X-energy is developing a report that documents reviews of these PIRTs to assess changes in the literature and state-of-the-art since they were performed, especially with regard to developments in TRISO technology development through the AGR Program.

3.4 ADDITIONAL GUIDANCE: ANS-53.1

The American Nuclear Society (ANS) Standard 53.1, “Nuclear Safety Criteria and Safety Design Process for Modular Helium-Cooled Reactor Plants” [ANS 2011] [50], was released in 2011 for trial use and revised in 2016. The standard has not been endorsed by the NRC, and as of 2020 was being balloted for extension of the trial use period while it is revised to align with NEI 18-04. Nevertheless, it provides a comprehensive approach for modular HTGR design and safety analysis that was developed by contributors with extensive gas-cooled reactor experience. Although NRC staff did not participate actively in the development of the standard, members of the staff did observe most of the meetings in which the standard was developed.

ANS-53.1 specifies that the specified maximum allowable coated fuel particle defect rate and heavy metal contamination rate from manufacture, operational failure rate, and accident condition failure rate shall be demonstrated by an acceptable fuel qualification program. The X-energy fuel qualification program described in this report is consistent with this standard.

3.5 U.S. HTGR PRECEDENTS

3.5.1 Peach Bottom 1

A construction permit was issued to Philadelphia Electric Company for the Peach Bottom Unit 1 HTGR plant in 1962, and a provisional operating license was issued in January 1965. This 40 MWe plant operated from 1967 to 1974 using HEU/thorium carbide BISO-coated particle fuel in the second batch-loaded

reactor core. The purged fuel element design used for this plant was very different from the later prismatic fuel elements, but it provided early experience in coated fuel particle fabrication and performance. Moreover, it served as a test bed for irradiating large quantities of early TRISO fuel particles in fuel test elements.

3.5.2 Fort St. Vrain

The Fort St. Vrain (FSV) Nuclear Generating Station was a prismatic fuel HTGR that generated 842 MW(t) to achieve a net output of 330 MW(e). FSV operated from 1974 to 1989. Licensing interactions on FSV were based on TRISO fuel containing thorium carbide in the fertile particles and mixed HEU uranium/thorium carbide in the fissile particles.

3.5.3 GASSAR

General Atomic Standard Safety Analysis Report (GASSAR) was General Atomics' standard safety analysis report for a 6-loop HTGR rated at 3000 MWt/1160 MWe [GASSAR 1975] [51]. The design closely approximated that considered by Philadelphia Electric Company for the Fulton Station. It was docketed by NRC for review as a standard plant. The TRISO fuel types analyzed in GASSAR were HEU uranium carbide TRISO fissile particles and thorium oxide BISO fertile particles. NRC issued an interim safety evaluation report documenting the status of the GASSAR review. The Fulton preliminary safety analysis report was reviewed as far as completion of the NRC safety evaluation report and ACRS review letter. These reviews indicated that at the time performance data and plateout information for a large HTGR were considered inadequate to determine an accurate model of FP release as required by 10 CFR 100. In its review of Fulton, the NRC staff employed a very conservative FP release model to circumvent these issues [Tokar 1976] [47].

3.5.4 MHTGR Conceptual Design Preliminary Safety Information Document Review

In 1989, the NRC published NUREG-1338, a draft preapplication safety evaluation report (PSER) [NRC 1989] [34], documenting the NRC staff's preapplication review of the General Atomics Modular High-Temperature Gas-cooled Reactor (MHTGR) design and the staff's conclusions from the review. The reference fuel particle for the MHTGR was TRISO-coated LEU UCO and was similar to the reference fuel for the Xe-100. Following the release of the draft of NUREG 1338, DOE submitted additional information for the fuel design in 1991 and 1992 and held two meetings with NRC staff on fuel design and fission-product transport in 1991. A draft of the final PSER was completed in December 1995 [NRC 1995] [38]. The 1995 draft of the final PSER is based on the draft PSER issued in 1989 and on a number of reports completed after the draft PSER was issued.

The 1995 draft final PSER confirmed the following overall conclusions of the 1989 draft PSER with respect to the fuel design, specifically:

- The NRC staff believes that fuel design and quality can be developed to meet the performance objectives proposed by DOE and required by the safety analyses but notes that this conclusion is dependent on the successful outcome of the research program; and

- The NRC staff notes that actual fuel performance in the Federal Republic of Germany (FRG) reactors, together with reported laboratory and in-pile tests, gives promise that fuel performance objectives can eventually be demonstrated.

However, NUREG-1338 also stated that the information provided for the modular HTGR up to that time had not demonstrated the necessary design and quality of fuel to meet these performance objectives. It identified the following information that the NRC staff assessed was necessary to reach a determination on the fuel:

- Design thicknesses of fuel particle coatings and the bases for these thicknesses given the proposed fuel failures from manufacturing, normal operation (neutron fluence), and accidents (temperature),
- Quality control of the manufacturing process for the fuel and resulting tolerances on the coatings,
- Fuel performance of specific coated particles and coating tolerances demonstrated from irradiation and safety tests ,
- Expected fuel temperatures throughout the core during accidents and the resulting volume-averaged failed fuel fraction, and
- Potential dose consequences shown to be within acceptable limits for the predicted volume-averaged failed fuel fraction.

NUREG-1338 also includes the following conclusions which will be considered in qualifying fuel for the Xe-100:

- The statistical question of how many fuel particles are needed in the irradiation and safety tests to justify the proposed low failed-fuel fraction within 95% certainty;
- The fuel design and containment proposed for the MHTGR, which NRC staff considered to be a licensability issue for the MHTGR. (Licensability issues occur when the design departs significantly from what NRC has accepted in the past or when changes in the design to resolve a staff concern could fundamentally alter the proposed design); and
- The credible mechanisms for “weak fuel” (fuel that performs acceptably during normal reactor operation but is subject to failure under more stringent conditions during accidents) to ensure that all mechanisms for fuel failure are recognized and quantitatively accounted for in the fuel-performance models.

NUREG-1338 states that the statistically low failure rates assumed in the fuel safety analysis will require a rigorous R&D program that complies with a systematic statistical approach commensurate with the number of parameters and the required accuracy. NUREG-1338 indicates that successful completion of the fuel R&D program must provide a statistically significant demonstration that:

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

The fuel design issues that were presented in the 1995 draft final PSER were taken into account in the development of the AGR Fuel Development and Qualification Program that was initiated at INL during the following decade (Section 5.3.2). Many of the other outstanding plant design issues noted in the 1995 draft final PSER were addressed during the NGNP project from 2006 to 2014 (Section 3.5.6).

3.5.5 Pebble Bed HTGRs

There have been several preapplication interactions with the NRC on pebble bed HTGR designs developed by other vendors/prospective licensees. In 2000, Exelon Generation Company began preapplication interactions with NRC on the feasibility of licensing the pebble bed reactor[5] design under the licensing provisions of 10 CFR 52. The pebble bed reactor featured a LEU UO_2 TRISO-coated particle fuel. NRC staff plans for the review were documented in SECY-01-0070 [NRC 2001] [39]. Attachment 2 of SECY-01-0070 outlined proposed pebble bed reactor preapplication activities.

Exelon ended the pebble bed reactor effort in April 2002. Preapplication reviews and related NRC policy resolutions activities were ended by September 2002. In late 2002, at Exelon's request, the NRC issued a close-out letter for this review. The letter stated that the NRC staff did not perform a detailed technical review of the documents that had been submitted and that the review that had been conducted was a limited screening review to ensure that the issues, review status, and views and positions noted within the documents were consistent within the NRC's views and understanding.

However, relative to qualification of fuel spheres, the NRC did issue a request for additional information to Exelon [NRC 2002] [124] which indicated that supplemental fuel qualification tests would be required to extend the German LEU UO_2 data base to make the full dataset acceptable for the selected licensing activities.

In 2006, the South African company PBMR (Pty) Ltd. submitted a series of pebble bed reactor preapplication papers to the NRC for review in support of their design certification process. However, at the request of PBMR (Pty) Ltd., these review activities were also ended early.

3.5.6 Next Generation Nuclear Plant (NGNP) Preapplication Review

In 2005, the U.S. DOE established the NGNP Project at the Idaho National Laboratory (INL). The project was intended to support near-term commercial deployment of a modular HTGR demonstration plant.

NGNP activities included development of a modular HTGR regulatory framework to support commercial deployment. The regulatory framework activities were jointly established by DOE and NRC and communicated to Congress in 2008 [DOE/NRC 2008] [40] and were closely coordinated under a memorandum of understanding between DOE and NRC. The approach to establishing the licensing strategy primarily focused on adapting existing nuclear power plant regulatory requirements to the needs of NGNP licensing and deployment. A summary of these licensing efforts is provided in [INL 2014a] [41].

[5] In this document, PBMR will refer solely to the South African company model. "Pebble bed reactor" with no abbreviation will be used for all other discussion

The NGNP Project systematically examined HTGR licensing precedents and NRC regulations as they relate to the NGNP safety case and associated plant design goals. In 2009, the project used this information to develop a strategic implementation plan for establishing the regulatory basis necessary to complete and submit a modular HTGR license application to NRC.

The safety design approach and licensing strategy of the NGNP centered on the RN retention capabilities of the TRISO-coated particle fuel instead of assuming significant fuel failure and relying on the reactor building to limit offsite exposure. This strategy—along with related HTGR design goals—aligned with NRC's Advanced Reactor Policy Statement regarding pursuit of less complex reactor designs with longer response time constants, passive reactor shutdown and passive heat removal with limited reliance on operator actions, minimization of severe accident potential, and providing multiple barriers to potential RN releases.

A key element of the NGNP licensing strategy was to document proposed approaches in a series of prelicensing white papers. Each white paper included a specific set of outcome objectives to support NGNP licensing and was developed with inputs from DOE and the NGNP Licensing Working Group. The NGNP Licensing Working Group included representatives from three domestic HTGR design firms, an owner-operator organization, and staff from INL's NGNP Regulatory Affairs team. Among the key white papers submitted to NRC for formal review and feedback were those on Fuel Qualification, Mechanistic Source Terms, and Licensing Basis Event Selection. The review and feedback process included extensive public meeting interactions, conference calls, and written correspondence focused on responding to NRC's requests for additional information.

Ensuing NGNP interactions resulted in an NRC staff working group drafting initial assessment reports in February, 2012, with proposed regulatory positions on the Fuel Qualification and Mechanistic Source Term White papers and on the Defense in Depth, Licensing Basis Event Selection, and Safety Classification of Structures, Systems, and Components white papers [NRC 2012a] [42]. In mid-2012, as the NRC continued its review of the NGNP white papers, NGNP's DOE/INL team and the NRC staff jointly identified and committed to focus forthcoming discussions on four key licensing framework topics. These four topics, documented in a letter from the project to the NRC [Petti 2012a] [52] were important areas of significant and longstanding regulatory uncertainty for the modular HTGR industry. The four key topical areas were:

1) Functional containment performance,
2) Licensing basis event selection,
3) Source terms, and
4) Emergency planning.

Fuel qualification was an important element of the functional containment performance topical area.

In March 2013, the NRC staff issued revisions of the initial assessment reports and a letter summarizing the staff feedback on the four topics in [Petti 2012a] [52]. The documents were submitted by the staff to the NRC's Advisory Committee on Reactor Safeguards (ACRS) and were reviewed in ACRS meetings in April and May, 2013. ACRS comments were provided to the NRC staff [ACRS 2013] [53], and the staff provided responses to those comments [NRC 2013] [54]. In the areas of fuel qualification and mechanistic source terms, the ACRS was supportive of the content of the staff assessment, the major findings of which are summarized below.

The NRC staff findings were subsequently updated, and the assessments of the four key topics and the Fuel Qualification and Mechanistic Source Terms were finalized and released in July 2014 [NRC 2014] [43]. A final assessment of the Defense in Depth, Licensing Basis Event Selection, and Safety Classification of Structures, Systems, and Components white papers was not released. Effectively, these key topics formed the basis for future licensing framework development evidenced in the guidance of NEI 18-04.

The NRC staff indicated that it found the NGNP-proposed performance standard concerning the modular HTGR functional containment to be reasonable. The functional containment approach limits RN releases to the environment by emphasizing retention of radionuclides at their source in the fuel rather than allowing significant fuel particle failures and subsequent reliance upon other external barriers to assure compliance with identified top level regulatory RN control requirements. This approach was further supported by the release of SECY-18-0096 and associated SRM.

The major finding pertinent to fuel qualification that was addressed in the NRC staff assessment reports was that the INL Advanced Gas Reactor (AGR) Fuel Program was determined by the NRC staff to be “reasonably complete within the context of pre-prototype fuel testing”. Early fuel test results “show promise in demonstrating much of the desired retention capabilities of the TRISO particle fuel”. Additional information from special tests in the first operating HTGR unit were determined to be most likely necessary to confirm that the coated particle fuel developed for NGNP retains fission products as predicted.

Ongoing AGR Fuel Program activities since the issuance of the NRC staff assessment reports for the NGNP white papers have continued to show results consistent with the anticipated radionuclide retention capabilities of TRISO-coated particle fuel. These results are presented in Section 5.3.2.

4 FUEL DESIGN AND PERFORMANCE REQUIREMENTS

This section defines the as-manufactured fuel quality requirements and the in-service fuel performance requirements for the Xe-100 reference fuel. The essential features of the Xe-100 reactor design that inform these in-service fuel performance requirements were presented in Section 2 [Mulder 2016] [12].

Section 5 provides a summary description of the international development of coated particle fuel leading to a common understanding of particle failure mechanisms and generally accepted design features to mitigate or eliminate these failure mechanisms such that the fuel requirements are met at the required confidence level. The conditions for the testing programs, which are defined in Section 6, are established to support a conservative performance envelope based on the demonstrated capability of the reference fuel, allowing flexibility in the evolving reactor design.

4.1. XE-100 TOP-LEVEL RADIONUCLIDE CONTROL REQUIREMENTS

Stringent, top-level radionuclide control requirements have been selected for the Xe-100 design. The design objective of setting the top-level RN requirements is to limit calculated dose for all LBEs so that regulatory requirements for protection of the health and safety of the public and protection of the environment are met at an Exclusion Area Boundary (EAB) that will be set only a few hundred meters from the reactor. A further design objective is to meet the Environmental Protection Agency (EPA) Protective Action Guide (PAG) limits at the EAB, thereby minimizing disturbances to the day-to-day activities of nearby members of the public. Limits on RN release from the reactor building that are consistent with these top-level RN control requirements are needed to establish the target values for the barriers to RN release and ultimately to establish allowable in-service fuel failure and as-manufactured fuel quality requirements (e.g., allowable heavy metal contamination, SiC coating defects, etc.). The Xe-100 design will also control RN releases in order to demonstrate protection of the workforce in accordance with 10 CFR 20.

A systems engineering protocol was developed within the MHTGR program in the 1980s for deriving in-reactor fuel performance requirements and as-manufactured fuel quality requirements from the top-level RN control requirements and was implemented during the NGNP Project as well [NGNP FQ WP 2010] [4]. A similar protocol is employed by the Xe-100 design beginning from the conceptual design and carried through each design baseline to manage the fuel requirements.

The logic for deriving these fuel requirements is illustrated in Figure 5. Top-level requirements are defined by both the regulators and the end user. Lower-level requirements are then systematically derived using a top-down functional analysis method. With this approach, the RN control requirements for each of the release barriers can be defined. For example, starting with the allowable doses at the site boundary, limits on Curie releases from the reactor building, reactor vessel, and reactor core can be successively derived. Fuel-failure criteria are, in turn, derived from the allowable core RN release limits. Finally, the required as-manufactured fuel attributes are derived from the in-reactor fuel-failure criteria, providing a logical basis for the fuel quality specifications.

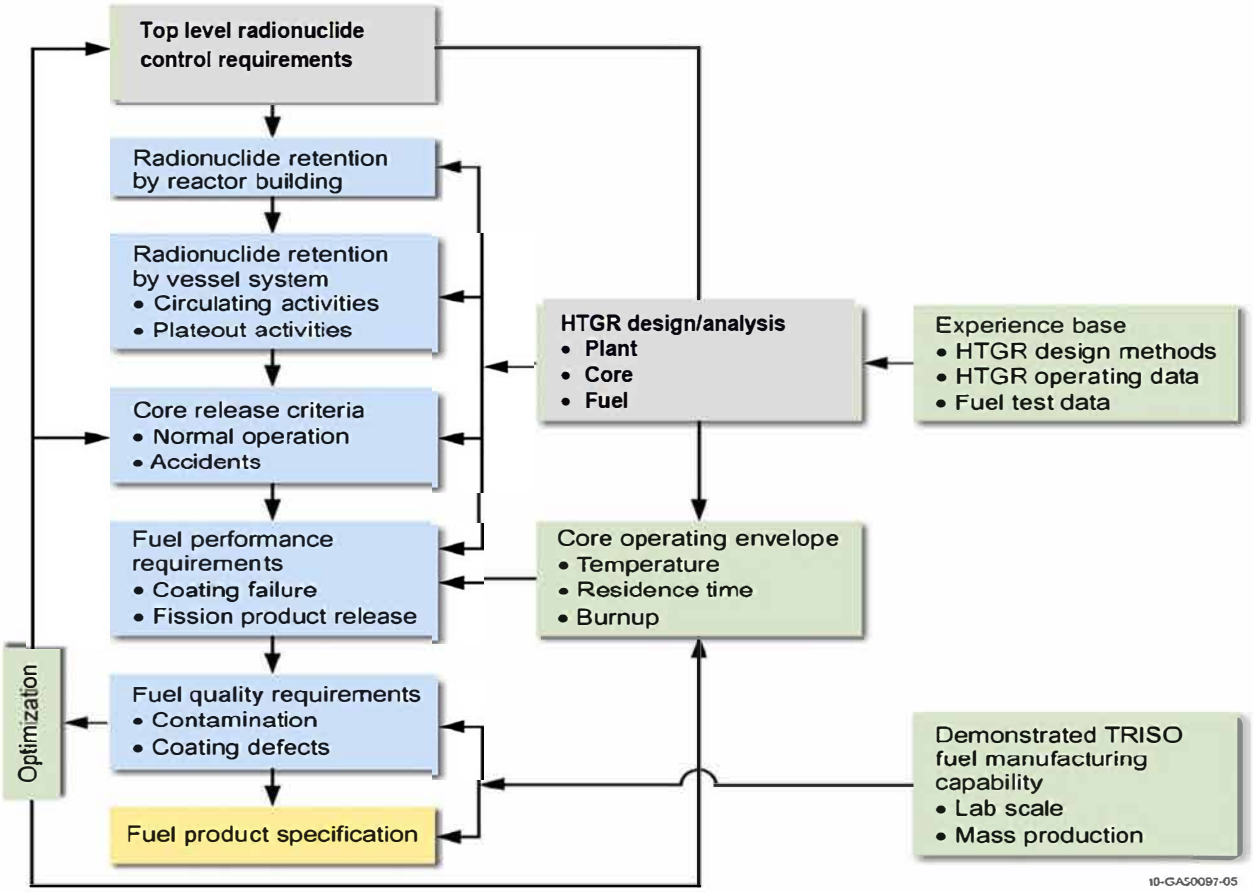

Figure 5: Logic for Deriving Fuel Product Specification

The key top-level radionuclide control requirements expected to be imposed for the Xe-100 design are listed in Table 2. The regulatory requirements are updated from those identified for the NGNP Project [NGNP MST WP 2010] [55].

Table 2. Key Top-Level Radionuclide Control Requirements for the Xe-100

Top Level Regulatory Requirements	
1	10 CFR 50, Appendix I, Limits for Radionuclides in Plant Effluents: • Whole Body Dose ≤ 5 mrem/yr • Thyroid Dose ≤ 15 mrem/yr
2	10 CFR 20 Subpart C Occupational Dose Limits: • Total effective dose equivalent (TEDE) ≤ 5 rem/yr • Organ Dose ≤ 50 rem/yr
3	10 CFR 20 Subpart D Public Dose Limits: • Annual TEDE ≤ 0.1 rem • Hourly External Dose ≤ 0.002 rem
4	40 CFR 190 Subpart B Environmental Standards: • Whole Body ≤ 25 mrem • Thyroid Dose ≤ 75 mrem • Organ Dose ≤ 25 mrem
5	10 CFR 52.47[a] Offsite Dose Limits for DBAs: • TEDE ≤ 25 rem for 2 hours at the EAB • TEDE ≤ 25 rem for the duration of the plume passage at the LPZ boundary
6	EPA PAGs for Radioactive Release for Public Sheltering & Evacuation and KI Administration [EPA 2017] [56]: • Early Phase Projected Dose 1 to 5 rem over four days • Early Phase Thyroid Dose 5 rem for potassium iodide administration • Intermediate Phase Projected Dose ≥ 2 rem in the first year; 0.5 rem/yr in the second and subsequent years
7	NRC Safety Risk Limits [NRC 1986] [57] • Quantitative Health Objectives (QHO) of individual risk of prompt fatality of 5×10^{-7}/ plant yr and of latent fatality of 2×10^{-6}/ plant yr. Evaluated over the area from the plant boundary to 1 mile for prompt and to 10 miles for latent • Overall assurance of negligible cumulative risks during normal operation and off-normal events
User Requirements	
1	PAGs at the EAB for Licensing Basis Events with a frequency ≥5×10^{-7}/ plant yr

[a] Similar requirements are included in 10 CFR 52.79 for combined licenses and in 10 CFR 50.34 for reactors licensed under Part 50.

It is expected that the User Requirement in the table above will likely be the most restrictive requirement for setting fuel performance and quality requirements for the Xe-100 design. In the late 1980s, meeting the PAGs at the EAB was determined to be the bounding RN control requirement for General Atomics (GA) 350 MWt steam cycle modular HTGR [PSID 1992] [17].

The preliminary as-manufactured fuel quality requirements and in-service performance requirements for prismatic modular HTGR fuel for the NGNP [NGNP FQ WP 2010] [4] are given in Table 3. The fuel requirements for the Xe-100 reactor are expected to be similar to those in the table or slightly less stringent due to the lower operating temperatures and lower reactor power level of the design. An initial fuel product specification for the Xe-100 reference fuel was prepared in "Xe-100 Reactor Fuel Specifications" [Mulder 2021a] [58] that is consistent with the requirements in Table 3 but the fuel requirements were taken from [Hanson 2009] [104] since the fuel quality and fuel performance requirements were not finalized for the Xe-100 at publication. This specification effectively uses the LEU UCO TRISO-coated particle specifications for the AGR-5/6/7 fuel qualification/margin test and the Siemens HTR-Modul specifications for the fuel sphere that were developed in the German pebble bed HTGR program in the late 1980s. The requirements for in-service performance are specified on a core-average basis. As described in the following section, the requirements for as-manufactured quality and in-service performance of coated-particle fuel have been based on a two-tier set of radionuclide design criteria, referred to as the "Design" and "Maximum Expected" criteria. The fuel quality and performance requirements of Table 3 are also presented in this two-tier format.

Table 3. Preliminary Xe-100 Fuel Requirements

Parameter	"Maximum Expected"	"Design"
As-Manufactured Fuel Quality:		
HM contamination	[[ ]]^{P}	$\leq 2.0 \times 10^{-5}$
Missing or defective buffer	[[ ]]^{P}	$\leq 2.0 \times 10^{-5}$
Missing or defective IPyC	[[ ]]^{P}	$\leq 1.0 \times 10^{-4}$
Defective SiC	[[ ]]^{P}	$\leq 1.0 \times 10^{-4}$
Missing or defective OPyC	[[ ]]^{P}	0.02
In-Service Fuel Failure:		
Normal operation	[[ ]]^{P}	$\leq 2.0 \times 10^{-4}$
Core heat-up accidents	[[ ]]^{P}	$\leq 6.0 \times 10^{-4}$

4.2. RADIONUCLIDE DESIGN CRITERIA

Standard design practice on the U.S. HTGR programs has been to define a two-tier set of RN design criteria—referred to as "Maximum Expected" and "Design" criteria (or allowable core releases for normal operation and AOOs). This practice has been followed since the design of the Peach Bottom 1 prototype

U.S. HTGR up through the NGNP prismatic SC-MHR [CDR 2010] [58]. The “Design” criteria are derived from externally imposed requirements or guidelines, such as site boundary dose limits, occupational exposure limits, EPA PAGs, etc. In principle, any of these RN control requirements or guidelines could be the most constraining for a given reactor design.

After the “Design” criteria have been derived from the RN control requirements, the corresponding “Maximum Expected” criteria are derived by dividing the “Design” criteria by a design margin, or “safety factor,” to account for uncertainties in the design methods. This factor has typically been a factor of four for the release of fission gases from the core and a factor of 10 for the release of fission metals [NGNP MST WP 2010] [55]. The fuel and core are to be designed such that there is at least a 50% probability that the FP release will be less than the “Maximum Expected” criteria and at least a 95% probability that the release will be less than the “Design” criteria. The approach to implementing such RN design criteria is illustrated in Figure 6 (no particular scale is implied in this figure; it is simply a conceptual illustration of the approach).

“Design” and “Maximum Expected” RN design criteria are used to determine the “Design” and “Maximum Expected” requirements for as-manufactured quality and in-service performance of coated-particle fuel for the Xe-100 design, as shown in Table 3.

In the example given in Figure 6, the preliminary design predictions (solid lines) exceed the criteria (double lines) at both the 50% and 95% confidence levels. The nominal (50% confident) prediction is slightly higher than the “Maximum Expected” criterion, and the upper bound (95% confident) prediction exceeds the “Design” criterion, because of large uncertainties in the predictive methods at this stage of the design. By the final design, the 50% confident prediction has been reduced to less than the “Maximum Expected” criterion by design optimization, and the 95% confident prediction has been reduced to less than the “Design” criterion by application of more accurate predictive methods resulting from technology development and methods validation.

In practice, an iterative procedure is required to develop optimized RN design criteria as the plant design matures during the conceptual, preliminary, and final design phases. With each iteration the lower level requirements are refined as a result of analyses performed for the reactor systems, structures, and components (SSCs). The goal is to produce an optimum design that meets all the top-level requirements with sufficient, but not excessive, margin.

The basis for use of a factor of four for the release of fission gases from the core and a factor of 10 for the release of fission metals as the design margin between the “Maximum Expected” criteria and the “Design” criteria in the U.S. HTGR programs is discussed in the MST white paper for the NGNP project [55] and X-energy’s MST licensing topical report [3].

A smaller design margin is specified for fission gas release (factor of four) than for fission metal release (factor of 10) because predicting gas release is less complex, and gas release is less sensitive to service conditions in the core, particularly to fuel temperatures. As described in X-energy’s MST licensing topical report [Loza 2021] [3], the dominant sources of fission gas release, which includes iodine and tellurium isotopes as well as noble gas isotopes, from a modular HTGR core are (1) as-manufactured HM contamination and (2) exposed fuel kernels. The HM contamination fraction is controlled by the fuel product specification, and the exposed kernel fraction is limited by the fuel and core designs. The exposed kernel fraction is calculated with fuel performance models derived from fuel testing, and the fractional

releases of fission gases from HM contamination and exposed kernels are calculated with experimentally determined release correlations.

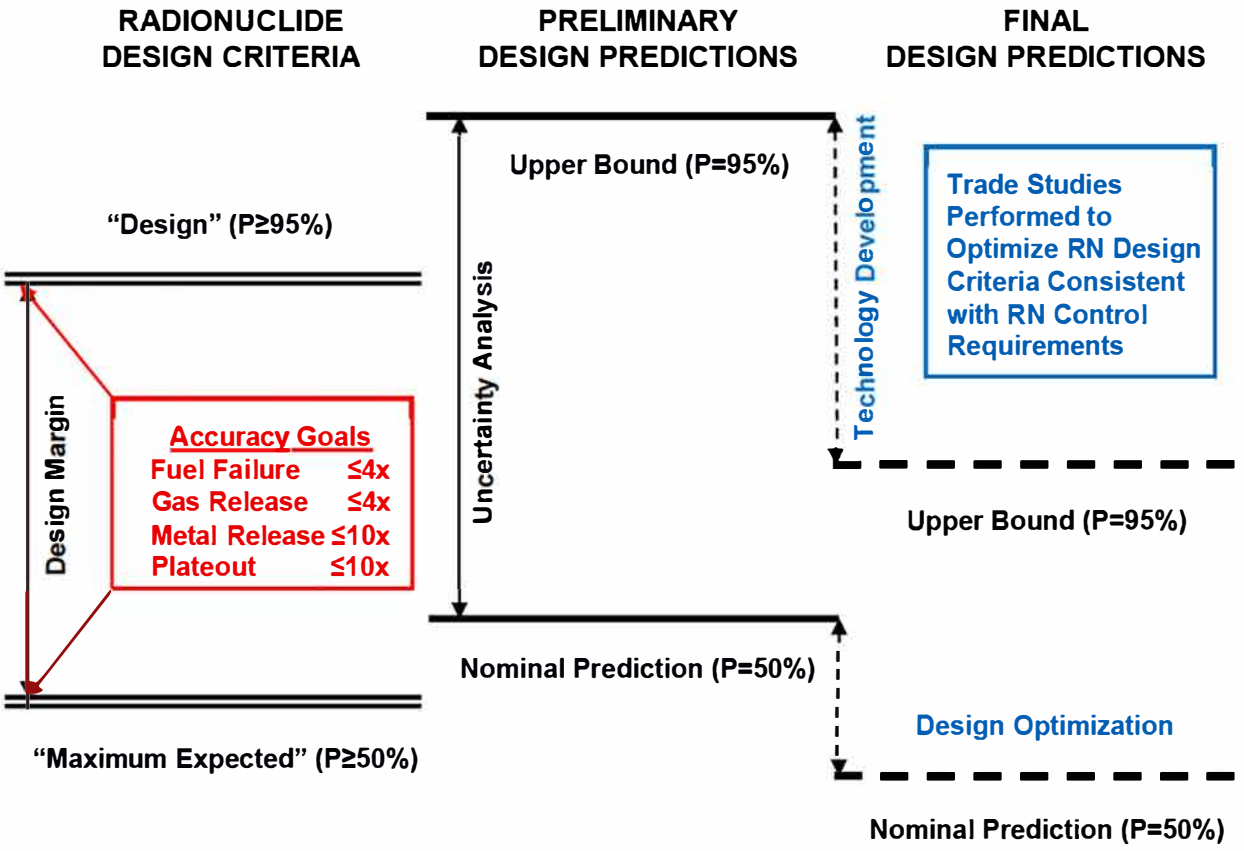

Figure 6: Radionuclide Design Criteria

Fission metal release from the core is more complex. In addition to metal release from HM contamination and exposed kernels, volatile metals (Ag, Cs, Sr, Eu) can also be released from "partially failed" particles with defective or failed SiC coatings but with at least one pyrocarbon coating intact. Moreover, Ag isotopes and, to a lesser extent, Sr and Eu isotopes can diffuse through intact TRISO coatings if the temperature is sufficiently high for a sufficiently long time. Once these volatile metals have been released from fuel particles, they must migrate through the fuel pebble matrix and finally into the flowing helium. In other words, there are many additional transport steps in the release of fission metals from the core as compared to fission gases. Moreover, the activation energies for metal transport in core materials are much larger than those for gas transport so that the former is more sensitive to uncertainties in predicted core temperature distributions.

While the current design margins are based largely upon engineering judgment and subject to refinement as the design and technology programs progress, they appear to be reasonable and attainable goals based upon the current knowledge base. As summarized in Section 4.6.1 of [NGNP MST WP 2010] [55] and elaborated in [TEC-978 1997] [69], [TEC-1645 2010] [70], and [TEC-1674 2012] [23]], a number of

comparisons have been made between code calculations and integral test data from operating HTGRs and in-pile tests to assess the accuracy of the design methods. In almost all cases, fission gas release and fission metal release have been calculated to well within factor of four and factor of ten accuracy goals, respectively.

4.3. XE-100 FUEL PRODUCT SPECIFICATION

The required as-manufactured attributes and properties of the LEU UCO fuel kernels, TRISO-coated particles, and fuel spheres containing those particles for use in the Xe-100 reactor core are defined and controlled by a fuel product specification. A draft fuel product specification for the Xe-100 is currently under development. While UCO TRISO-coated particles have been under development for decades, the AGR program has significantly refined and optimized the processes for fabricating them and has demonstrated their superior performance by completing a series of irradiation tests and post irradiation heating (PIH) tests (Section 5.3.2). The attendant optimized specifications are captured in the fuel product specifications for the AGR-5/6/7 fuel qualification/margin test [Marshall 2016] [106], and these specifications have been preliminarily adopted for the Xe-100 UCO fuel kernels and TRISO-coated particles.

The AGR program is focusing on UCO TRISO-coated particles in cylindrical fuel compacts; therefore, the preliminary fuel spheres specifications for the Xe-100 will be primarily based upon the fuel product specifications for the German HTR Modul [HOBEG 1989] [60] and the Chinese HTR-PM [Zhou 2013] [10], which are both modular pebble-bed designs with fuel requirements similar to the NGNP requirements.

Production of high-quality TRISO-coated particle fuel is achieved through a combination of product specifications, QC methods, and manufacturing equipment specifications, each of which plays an important role.

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While examples of fuel product specification information are provided herein, it is expected that each licensing application will include its associated fuel specification as part of the licensing bases in order to demonstrate the alignment of fuel system design bases, fuel performance criteria, the modeling of fuel performance for transient and safety analyses, and the associated mechanistic source term calculations and evaluation of LBEs against their associated acceptance criteria.

5 COATED-PARTICLE FUEL EXPERIENCE BASE

This section addresses the existing experience base that supports the development, qualification, and production of TRISO-coated particle fuel for the Xe-100 modular HTGR and serves as the historical basis for the reference fuel design of UCO TRISO in spheres. A broad base of experience encompassing a range of coated particle designs and service conditions has produced a general understanding of the important phenomena associated with particle fabrication and performance and has served to identify potential fuel failure mechanisms. This experience led to a common international set of particle design features that, in combination with restrictions on service conditions, mitigate or eliminate failure mechanisms. The successful German experience with the performance of LEU UO_2 pebble-bed fuel provides general support for all TRISO-coated particle fuel designs. The superior performance of UCO TRISO fuel demonstrated by the AGR program provides the technical justification for its selection as the reference particle for the Xe-100 reactor.

Section 5.1 summarizes the broad international experience with coated-particle fuel covering a wide range of particle designs explored in the evolution toward the TRISO-coated particle design under common development today. It also addresses the failure mechanisms that have been identified from this experience. Of particular importance, Section 5.2.1 summarizes the fabrication, irradiation, safety testing, and analysis methods experience from the German pebble fuel development program that directly applies to the Xe-100 design. Section 5.3 summarizes the U.S. prismatic fuel fabrication, irradiation, safety testing, and analysis methods experience that directly applies to the LEU UCO particle design. The activities planned to support the qualification of the reference Xe-100 fuel are addressed in Section 6.

5.1. BASIC CONSIDERATIONS

Experience with manufacturing coated-particle fuel has demonstrated the feasibility of producing large quantities of TRISO fuel with low as-manufactured defect levels (approaching defect fractions of 10^{-5}). This was first demonstrated in Germany, with the fabrication of LEU UO_2 reload fuel batches for the Arbeitsgemeinschaft Versuchsreaktor (AVR) [TEC-978-1997 [69] and TEC-1645-2010 [70]] and subsequently confirmed in fuel fabrication campaigns in Japan for the High Temperature Test Reactor (HTTR) first core and in China for the 10 MW(t) High Temperature Gas-Cooled Reactor (HTR-10) first core [Zhao 2006] [61] and the ongoing fuel fabrication campaign for the HTR-PM commercial modular HTGRs [Zhou 2013] [10]. During the past decade, high-quality UCO TRISO-coated particle fuel in compacts has been fabricated by the AGR program, and this UCO TRISO fuel has been demonstrated to have a higher burnup and higher temperature capability than LEU UO_2 TRISO fuel in safety tests performed by the AGR program [Petti 2014] [63] as elaborated in Section 5.3.2.

5.1.1 Evolution of Coated-Particle Fuel

This section addresses the broad range of existing coated-particle fuel experience that contributes to the general understanding of coated-particle fuel fabrication and performance. This general experience includes a wide range of kernel and coating properties. The data summarized in the following sections are documented in international publications produced by national programs over several decades.

By the early 1960s, coated-particle fuel development for graphite-moderated helium-cooled HTGRs was well under way in the United Kingdom in support of the DRAGON research reactor [Simon 2002] [13], in the U.S. in support of the Peach Bottom Unit 1 prototype power reactor [INL 2003] [109], and in Germany in support of the AVR research and power reactor [VDI-Verlag 1990] [24]. Coated particle designs for these reactors varied considerably, as illustrated in Figure 7 (the AVR fuel evolved through many designs in the course of over two decades of plant operation, including the LEU UO_2 TRISO design discussed in Section 5.1.2).

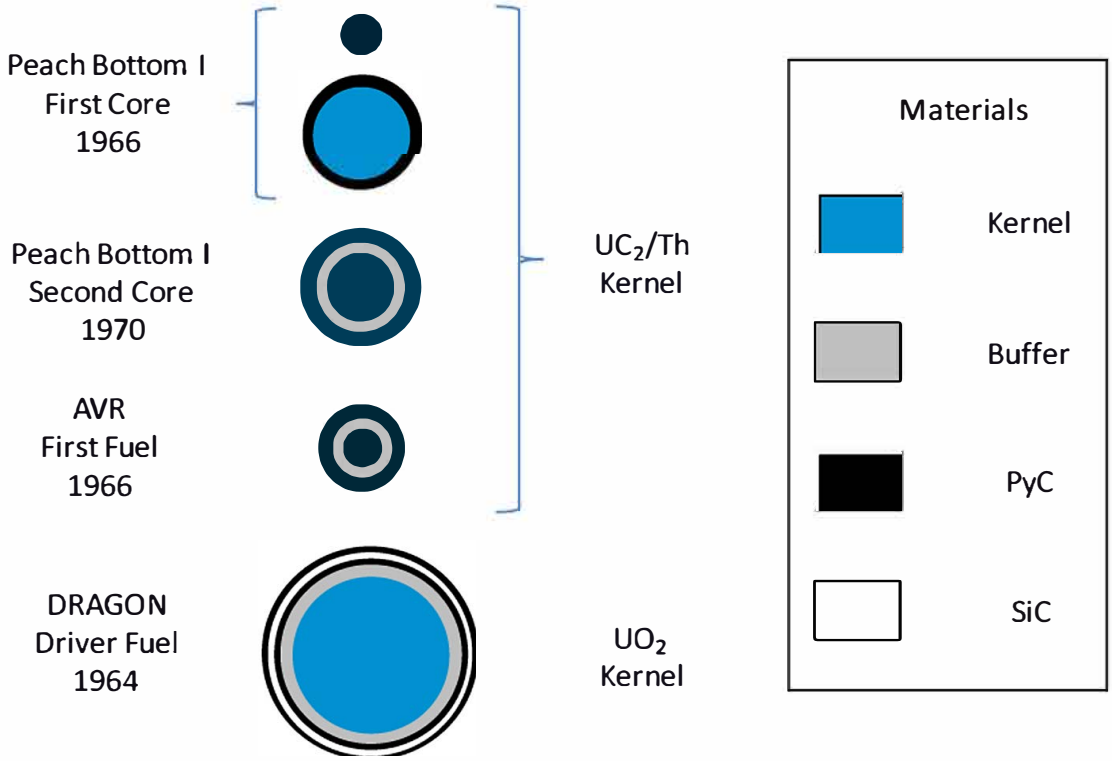

Figure 7: Early Coated-Particle Designs

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

- Kernel characteristics
  - Diameter – 100 to 800 µm
  - Fissile/fertile materials – uranium, thorium, plutonium (mixed and unmixed)
  - Chemical forms – oxide, carbide, oxycarbide
  - Enrichment – ranging from depleted to highly enriched uranium and plutonium
- Coating characteristics
  - BISO – variations in buffer and PyC coating thicknesses and properties

  - TRISO – variations in buffer, PyC and SiC (or zirconium carbide) thicknesses and properties

- Fuel forms
  - Spheres – multiple sizes (e.g., fuel-free zone thickness, etc.) and fabrication methods
  - Compacts – cylindrical and annular shapes with variations in particle packing fractions and fabrication methods
- Irradiation facilities
  - Materials Test Reactors – HFR (Netherlands), FRJ 2 DIDO (Germany), IVV-2M (Russia), SILOÉ (France), R2 (Sweden), BR2 (Belgium), High-Flux Isotope Reactor (HFIR) (U.S.), Advanced Test Reactor (ATR) (U.S.), with variations in spectra and the degree of irradiation acceleration
  - Research and Demonstration Reactors – DRAGON (United Kingdom), Peach Bottom I (U.S.), AVR (Germany), FSV (U.S.), Thorium High Temperature Reactor (THTR) (Germany), HTTR (Japan), HTR-10 (China), HTR-PM (China).
- Irradiation and testing conditions
  - Burnup – ranging from >1% to <70% fissions per initial metal atom (FIMA)
  - Fast fluence – ranging from $>1 \times 10^{21}$ to $<10 \times 10^{21}$ n/cm2
  - Irradiation temperature – ranging from 600 to 1,950°C
  - Accident simulation temperature – ranging from 1,400 to 2,500°C.

This broad range of experience and data has resulted a fairly detailed understanding of the parameters and phenomena of importance in the fabrication and performance of coated-particle fuel. The international experience base is summarized in three IAEA Technical Documents [TEC-978 1997 [69], TEC-1645 2010 [70], and TEC-1674 2012 [23]]. In considering this experience and data, the international community has largely converged on common LEU TRISO-coated particle designs as discussed in Section 5.1.2, having very similar coating thicknesses and properties with variations in kernel diameter, enrichment, and composition (UO_2 and UCO), depending on the specific service conditions and requirements.

5.1.2 TRISO-Coated Particle Design

The broad coated-particle fuel fabrication, irradiation, and testing experience introduced in Section 5.1.1, combined with effective international information exchanges, especially through the IAEA, has resulted in a consensus on basic coated-particle properties among ongoing fuel-development programs, as illustrated in Figure 8 and discussed below. This common coated-particle design approach mitigates or eliminates the failure mechanisms discussed in Section 5.1.4 and incorporates the following elements:

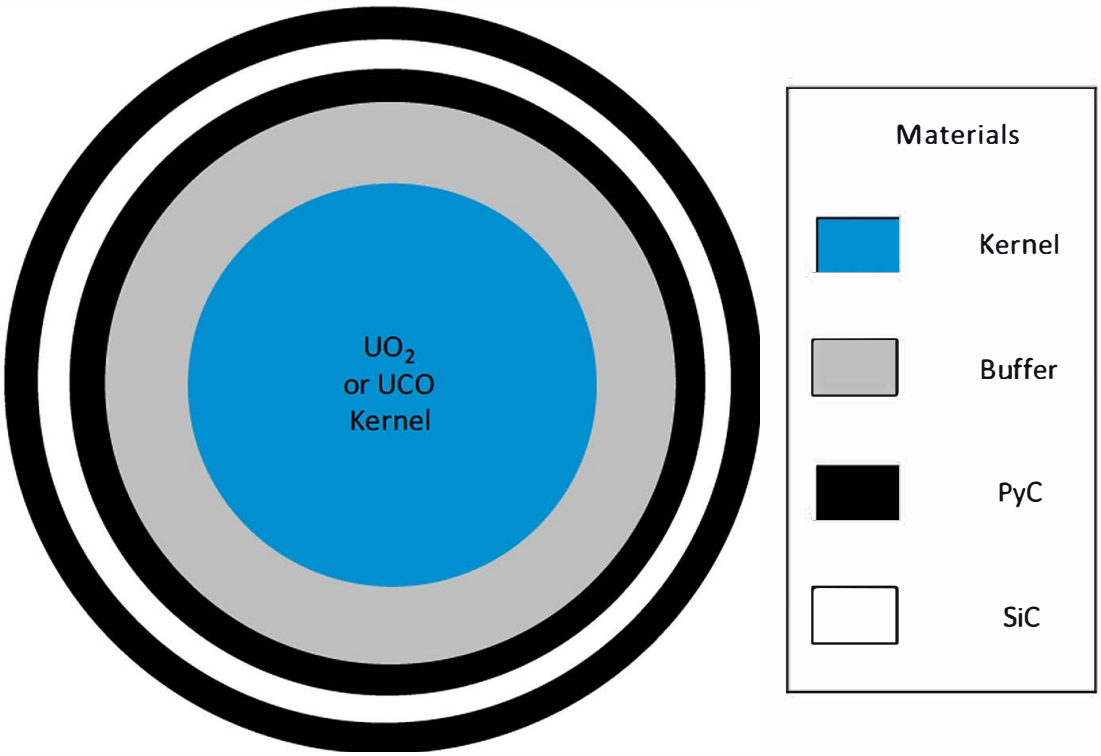

Figure 8: The International-Consensus TRISO-Coated Particle Design.

- *Kernel* – The fuel kernel consists of high-density, low-enriched (<20% ^{235}U) UO_2 or UCO.[6] The kernel serves as an important barrier to RN release by immobilizing many of the fission products and delaying the release of others. The UO_2 kernel performs effectively within the range of burnup and temperature gradients experienced in the German pebble-bed designs. The UCO kernel functions as getter of excess oxygen produced by fission, limiting the generation of CO and CO_2 and associated increased gas pressure in the particle and kernel migration (Section 5.1.3), thus allowing higher burnup limits and thermal gradients associated with U.S. prismatic designs.
- *Buffer* – The buffer layer consists of a low-density (~50% of theoretical density) isotropic pyrocarbon. The primary purpose of the buffer layer is to provide void volume for gaseous fission products to limit pressure buildup within the coated particle. As a compressible material, it serves to mechanically decouple the kernel from the inner pyrocarbon layer to accommodate kernel swelling, thereby reducing the buildup of stress in the outer coating layers during irradiation. The buffer layer also absorbs energetic fission products recoiling from near the kernel surface, thus protecting the inner pyrocarbon layer of the coated particle.
- *Inner Pyrocarbon Coating* – The inner high-density isotropic layer of pyrocarbon (IPyC) forms the first load-bearing barrier against the pressure exerted by gaseous fission products and reaction products (CO, CO_2) within the fuel kernel and buffer layer. The IPyC layer also serves to protect the kernel from corrosive gases (HCl, Cl_2) liberated during the SiC coating process. Both the IPyC and OPyC layers retain

[6]Additional coated-particle fuel development using kernels containing other actinides, primarily plutonium and neptunium, has also been evaluated as a means of reducing long-lived radionuclides contained in spent fuel.

gaseous fission products but become less effective in retaining metallic fission products at higher temperatures. The anisotropy of the IPyC layer is limited to control dimensional changes during irradiation, during which the IPyC and OPyC layers shrink at first but may expand again if sufficiently high fast-neutron dose levels are reached. The interaction between the IPyC and OPyC high-density pyrocarbon layers and the SiC layer sandwiched between them plays an important role in keeping the SiC layer under compressive stress during irradiation.

- *Silicon Carbide* – As noted above, the IPyC and OPyC layers become less effective in retaining metallic fission products at higher temperatures. A primary purpose of the SiC layer is to prevent the release of these mobile fission metals from the reactor core. While the SiC layer has sufficient strength to withstand internal pressure produced during irradiation, TRISO-coated particles are designed to keep the SiC in compression during irradiation to design burnups.
- *Outer Pyrocarbon Coating* – The primary function of the OPyC layer is to protect the SiC layer against damage in the fuel manufacturing processes following the coating process. It also provides pre-stress on the outside of the SiC because of its net shrinkage under fast-neutron irradiation—thereby reducing the tensile stress in the SiC layer—and serves as a redundant barrier to gaseous FP product release. The anisotropy of the OPyC layer is limited to control dimensional changes during irradiation.

5.1.3 Kernel Composition

Coated-fuel particles with a variety of kernel compositions, including UO_2, UCO, UC_2, ThO_2, ThC_2, $(U,Th)O_2$, $(U,Th)C_2$, PuO_{2-x} and $(Th,Pu)O_{2-x}$, and enrichments have been fabricated and tested.

Selection of the kernel composition is a tradeoff since each type has distinct advantages and disadvantages. Carbide-based kernels, which were used in Peach Bottom, Fort St. Vrain and initially in AVR, permit high burnups and are thermally stable; however, they readily hydrolyze when exposed to water with increased fission gas release, and they do not retain fission metals well. Oxide-based fuel kernels, such as UO_2, are (relatively) easy to fabricate, best retain metallic fission products, and are most resistant to corrosion by water or air. However, with pure UO_2, CO is formed at higher burnups, which contributes significantly to the internal gas pressure of the particle. In fact, this thermochemical behavior has been largely confirmed by measuring the CO content of irradiated UO_2 particles [Proksch 1982] [111]. Perhaps most importantly, CO can corrode the SiC layer at accident temperatures if there are through cracks in the IPyC [Minato 1972] [64] and [Morris 2016] [62]; UO_2 also has the greatest potential for kernel migration (a failure mechanism described in the next section).

These disadvantages of UO_2 can be eliminated by including a fraction of UC_2 in the kernel with the optimal proportion of UC_2 being dependent the maximum design burnup. Given the importance of the kernel chemistry on the performance of coated-particle fuel, it is discussed in greater detail below.

The solid-state phase equilibria in the U-C-O system are a major controlling parameter in the production of the fuel kernel (especially for UCO kernels) and are important to its performance during normal operation and under most accident conditions. The phase relationships in the U-C-O system have been studied [e.g., Javed 1970 [65] and Potter 1972 [66]]; Appendix A of [NPR Fuel TDP 1992] [67] provides an excellent summary of the available thermochemical information; however, this reference is not generally available so it is excerpted here.

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Figure 9: [[ ]]^{P}

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[7]The U-C-O phase diagram is also applicable to pure UO_2 kernels because the oxide kernel is surrounded by carbon in a closed chemical system; however, with dense UO_2 kernels the time to approach equilibrium will probably be longer than for UCO kernels wherein the oxide and carbide phases are intimately mixed.

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With stoichiometric oxide fuel particles, including UO_2, carbon monoxide is produced from excess oxygen liberated upon fissioning of the heavy metal because the fission products in the aggregate are thermochemically incapable of binding all of the oxygen.

One effective way to control the CO pressure within uranium fuel particles is to maintain carbides within the kernel that can be oxidized in preference to elemental carbon. Each ^{235}U fission within UO_2 leads to fission products that, at maximum (assuming the formation of oxides of yttrium, cerium, lanthanum, neodymium, praseodymium, samarium, zirconium, strontium, europium, and barium), may combine with only ~1.62 of the two oxygen atoms released, leaving, at a minimum, 0.38 atoms available to oxidize other materials, such as carbon or carbides.

The equilibria of uranium and FP oxides with their carbides, plotted as a function of temperature in Figure 10 [Homan 1977] [71], show the stabilities of the oxides relative to one another and to CO. In this plot, the equilibria at the lowest oxygen potentials (the lowest lines) represent the most stable oxides (least stable carbides). That is, the rare earths yttrium, cerium, lanthanum, neodymium, and praseodymium are the most stable oxides, whereas barium is the least stable FP oxide (the most stable FP carbide) on the plot. Note that SiC is very stable. The equilibrium among C, CO_2 and CO has a negative slope with temperature, and therefore the oxidation of carbon to CO is more favored than the oxidation of BaC_2 to BaO above 1700°K and more favored than the oxidation of ZrC to ZrO_2 above 1900°K.

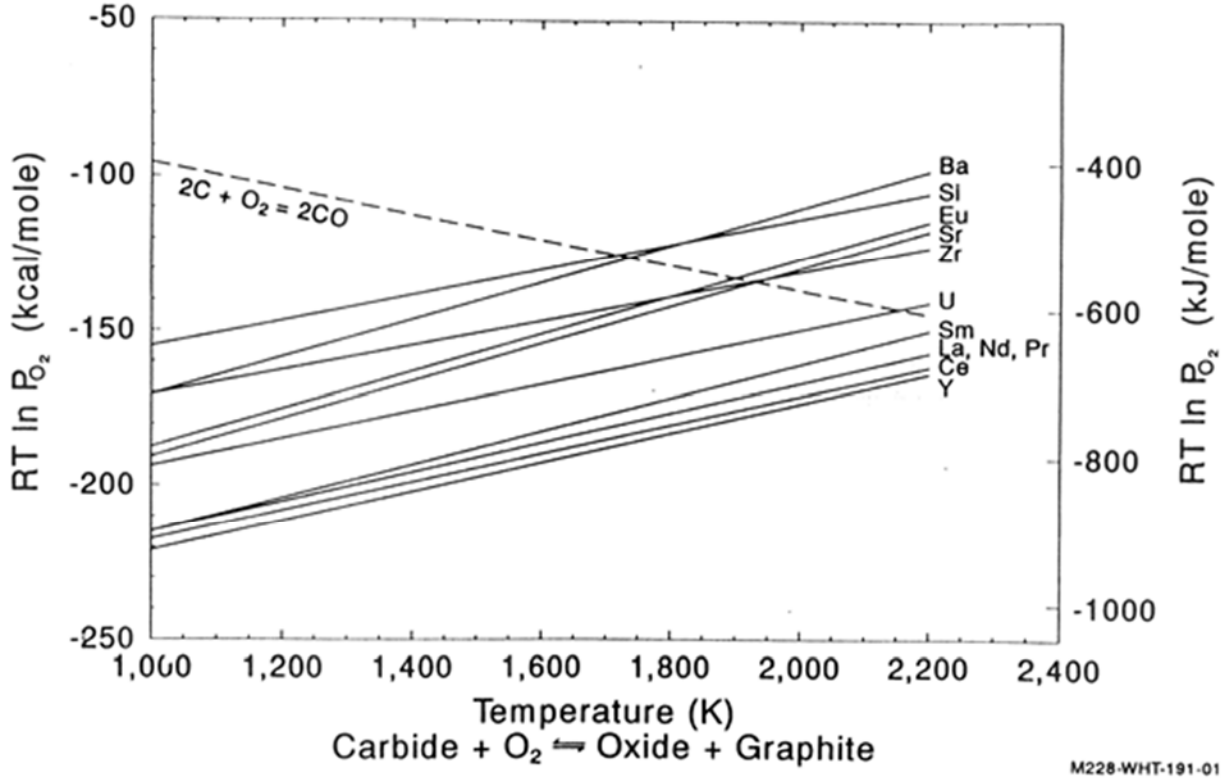

Figure 10: Oxygen Potential vs. Temperature for Various Oxide-Carbide Equilibria

Oxycarbide fuel is designed such that UC_2 is converted to UO_2 from the reaction with O_2 liberated by fissioning of UO_2. The oxygen potential is fixed by the UC_2/UO_2 equilibrium, meaning that rare-earth fission products will form oxides and the fission products zirconium, strontium, europium, and barium will form carbides. After exhaustion of UC_2, the oxygen potential in the kernel will shift upwards to the ZrC/ZrO_2 equilibrium. Figure 10 indicates that CO can be formed at temperatures less than 1700°K only after total conversion of BaC_2 to BaO. Note that SiC oxidation to SiO_2 can occur at oxygen potentials close to the BaC_2/BaO equilibrium.

5.1.4 TRISO-Coated Particle Failure Mechanisms

The primary challenges to TRISO fuel particle performance are phenomena that are characterized by extended times at elevated temperatures. Phenomena in LWRs that can lead to rapid changes in local conditions or in fuel integrity (e.g., departure from nucleate boiling, fuel/coolant chemical reactions) are precluded by the modular HTGR reactor characteristics discussed in Section 2.4.3. There are no transient perturbations that can lead to sudden initiation and propagation of fuel failures. Thus, the reactor design emphasis is on limiting the exposure of the fuel to elevated temperatures for extended periods of time.

The following failure mechanisms have been identified as capable of causing functional degradation or through-coating failure of the TRISO-coating system under irradiation and/or during postulated accidents [e.g., Petti 2012b [72] and TEC-1674 2012 [23]]:

- Pressure-vessel failure of standard (“intact”) particles (i.e., particles without manufacturing defects)
- Pressure-vessel failure of particles with defective or missing coatings
- Irradiation-induced failure of the OPyC coating
- Irradiation-induced failure of the IPyC coating and potential SiC cracking as a result
- Failure of the SiC coating caused by kernel migration in the presence of a temperature gradient
- Failure of the SiC coating caused by fission-product/SiC interactions
- Failure of the SiC coating caused by CO/SiC interactions
- Failure of the SiC coating resulting from thermal decomposition
- Failure of the SiC coating caused by HM dispersion in the buffer and IPyC coating layers.

All of these fuel failure mechanisms can be limited or “engineered out” of the fuel system through a combination of fuel kernel choice and careful design of the reactor core and its service condition envelope. As discussed in Section 5.3, the AGR program has confirmed that UCO TRISO-coated fuel particles can be shown to perform, with margin, in a manner consistent with the safety design approach of modular HTGRs.

These failure mechanisms are shown schematically in Figure 11. Phenomenological performance models, typically inspired by first principles and correlated with experimental data, have been developed to model each of these failure mechanisms [e.g., Martin 1993] [73].

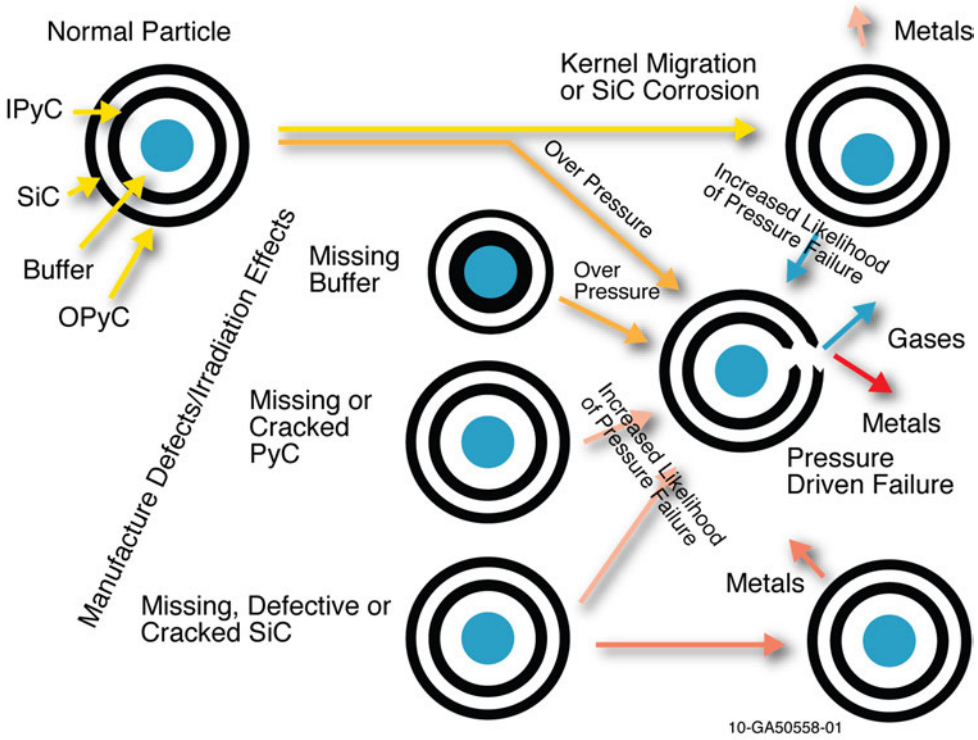

Figure 11: TRISO-Coated Particle Failure Mechanisms.

As-manufactured HM contamination is not an in-service degradation or failure mechanism but is very important with respect to fission product release. It is an extreme case of as-manufactured coating defects, whereby trace amounts of heavy metal are not encapsulated by a single intact coating layer (analogous to "tramp uranium" in LWR fuel). Modern fuel-product specifications only allow small fractions of HM contamination (~10^{-5} is typical); nevertheless, it is an important source of fission product release, especially of short-lived fission gases, including radiologically important iodine isotopes.

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

5.1.4.1 Structural/Mechanical Mechanisms

During irradiation, long-lived and stable fission gases are released from the kernel into the buffer, which increases the internal gas pressure. For some particle designs (e.g., UO_2 TRISO), carbon monoxide can also be generated during irradiation, which further increases the gas pressure. Because the SiC layer has a much higher elastic modulus than the PyC layers,[8] it would bear most of the internal pressure, which would produce a tensile stress if the irradiation-induced dimensional changes of the PyC and SiC layers were comparable. However, the PyC layers undergo shrinkage during irradiation, which produce

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

compressive stresses in the SiC layer. Within the range of allowed fuel-service conditions (e.g., temperature and fast neutron fluence), the compressive forces from PyC shrinkage more than compensate for the tensile stresses from internal pressure, such that the SiC remains in compression provided at least one of the PyC layers remains intact.

5.1.4.1.1 PyC Performance

As discussed above, shrinkage of the PyC layers during irradiation is a favorable attribute in terms of the compressive forces induced in the SiC layer. However, PyC shrinkage produces tensile stresses in the PyC layers themselves, which can lead to failure of these layers. The strains and stresses generated in the PyC layers are complex functions of fast-neutron fluence, irradiation temperature, and coating-material properties.

A property that greatly affects PyC performance is anisotropy, which can be quantified using x-ray or optical diffraction techniques. Anisotropy is usually expressed in terms of the Bacon Anisotropy Factor (BAF). For a perfectly isotropic material, BAF = 1, and for a perfectly oriented medium, BAF = ∞. Sufficiently isotropic PyC layers (BAF_o ≤1.035) are able to perform well out to high fast-neutron fluences because irradiation-induced creep relaxes the irradiation-induced strains and stresses to some extent.

5.1.4.1.2 Irradiation Induced Failure of IPyC Leading to SiC Cracking

Post-irradiation examination of fuel from the HRB-21 irradiation and the NP-MHTGR irradiations coupled with mechanical analyses showed that fuel particle failures in these irradiation experiments were caused by irradiation induced failure (cracking) of the highly anisotropic IPyC leading to tensile stress intensification in the adjacent well-bonded SiC layer causing subsequent cracking of the SiC layer [Leikind 1993 [74] and Hobbins 1993 [75]]. These failure analyses led to changes in the coating conditions used in the fabrication of fuel particles in the NGNP/AGR Fuel Program to ensure IPyC coatings with adequate isotropy were produced.

Instead of the classical radial cracking of the IPyC as a result of the tensile stress exceeding the ultimate tensile strength of the IPyC, a complex structural failure mechanism involving partial debonding of the buffer from the IPyC has been observed in the PIEs of AGR-1 and AGR-2 [Petti 2016] [21].

5.1.4.1.3 Pressure-Vessel Failure

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

$$\sigma_{SiC} = \frac{P r_{SiC}}{2 t_{SiC}} \tag{1}$$

where

P = internal pressure inside the particle

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

t_{SiC} = thickness of the SiC layer.

Pressure vessel failure occurs when the tensile stress in the SiC layer exceeds the strength of the SiC layer. The fraction of particles with a failed SiC coating[9], f_{SiC}, is calculated using Weibull statistical strength theory, assuming volume flaws and a uniform stress distribution in the SiC layer as:

$$f_{SiC} = 1 - \exp\left[-\left(\frac{\sigma_{SiC}}{\sigma_o}\right)^m V_{SiC}\right] \quad (2)$$

where

σ_o = Weibull characteristic strength

m = Weibull modulus

V_{SiC} = volume of the SiC layer.

5.1.4.2 Thermochemical Mechanisms

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

5.1.4.2.1 Kernel Migration

Local fuel temperatures and temperature gradients across the fuel can be relatively high when the reactor is producing power, especially in a prismatic core. Under these conditions, oxide and carbide fuel kernels can migrate up the thermal gradient. This kernel migration (KM) phenomenon is often referred to as the "amoeba effect" and can lead to complete failure of the coating system [W-L 1977] [76]. For oxide kernels, migration may be caused by carbon diffusion or gas phase diffusion of CO or other gaseous carbon compounds. An example of KM failure is shown in Figure 12. Failure by this mechanism is correlated as a function of temperature, thermal gradient, and thicknesses of the buffer and IPyC layers. Failure is assumed to occur when the kernel material contacts the SiC layer. The particle-to-particle variations in the buffer and IPyC thicknesses (expressed as normal distributions with measured variances) are accounted for when calculating the failure probability. Since CO formation is suppressed with UCO fuel, this failure mechanism is effectively precluded for the Xe-100 reference fuel, as confirmed by the results of the AGR Program to date (Section 5.3.2).

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

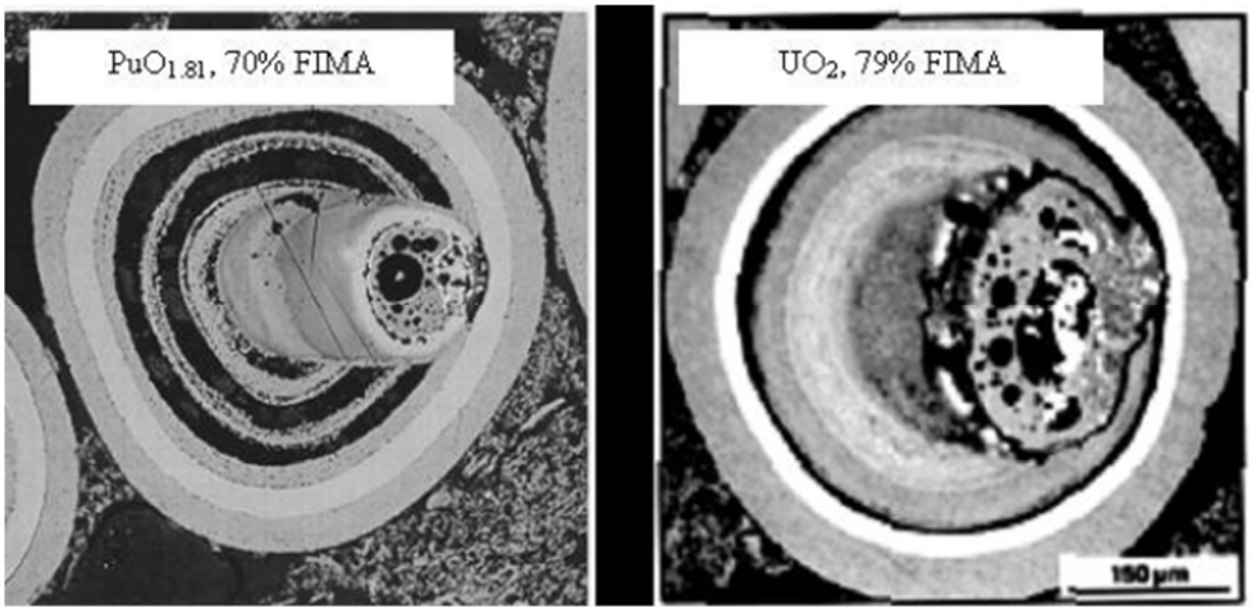

Figure 12: Kernel Migration in High-Burnup Oxide Fuels

5.1.4.2.2 Chemical Attack of SiC

Noble metals (e.g., Ru, Rh, Pd, and Ag) are produced with relatively high yield during fission of uranium fuels (the fission yields are highest in bred plutonium). During irradiation, the thermochemical conditions are not conducive for these elements to form stable oxides, and they can readily migrate out of the fuel kernel, regardless of its composition. Reactions of SiC with Pd have been observed during PIE of TRISO fuel [e.g., Ketterer 1985] [77]. Although the quantity of Pd is small compared with the mass of the SiC layer, the reaction is highly localized, and complete penetration of the SiC layer can occur if high temperatures are maintained for a sufficient period of time (see Figure 13). The reaction rate is highly dependent on temperature, so the time required to penetrate the SiC layer decreases rapidly as the temperature increases above about 1400°C [[

]]^{P}

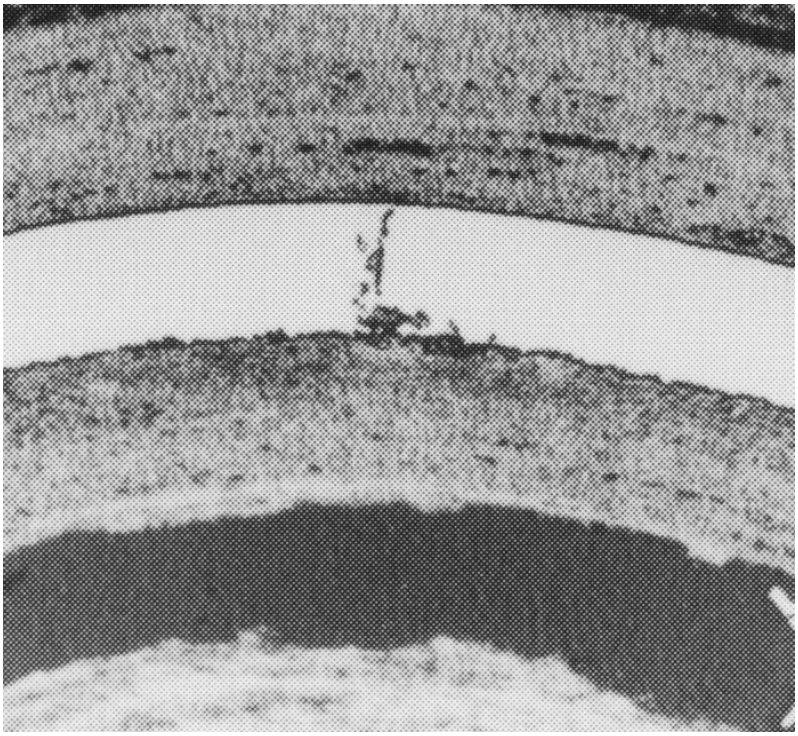

Figure 13: Localized Fission-Product Attack of the SiC layer.

Chemical attack of the SiC layer by CO has been observed in UO_2 particles irradiated at temperatures above approximately 1,400°C [Minato 1972] [64]. As shown in Figure 14, SiC degradation occurs near locations where the IPyC layer has cracked. The kernels of particles with degraded SiC layers were examined with an electron microprobe, which showed the presence of silicon in the form of fission-product silicides. [[

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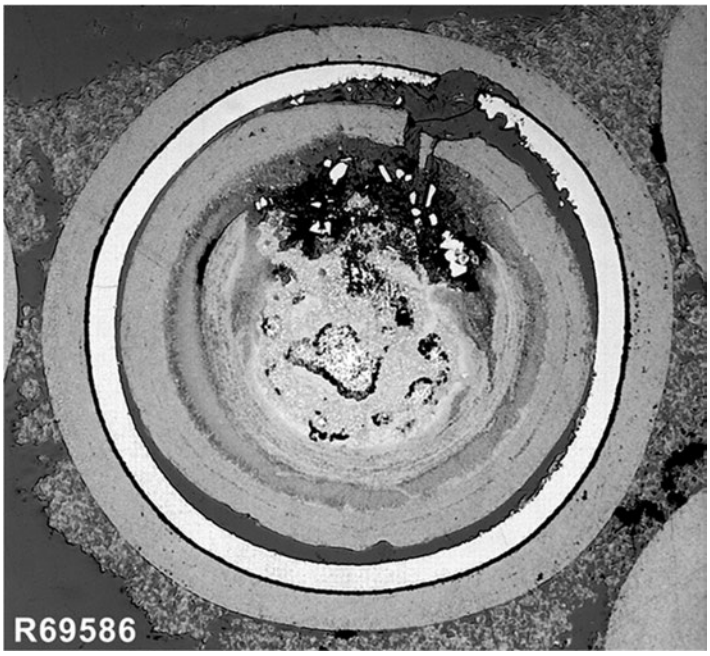

Figure 14: CO Corrosion of the SiC Coating in LEU UO_2 TRISO Fuel

5.1.4.2.3 Thermal Decomposition of the SiC Layer

At very high temperatures, SiC will decompose into its constituent elements. The silicon vaporizes, leaving a porous carbon structure as shown in Figure 15. Note that the particle does not experience catastrophic pressure-vessel failure even after the SiC coating is completely decomposed. Rather the coatings become porous and the fission gases (and also CO in the case of UO_2) permeate through the pyrocarbon coatings (i.e., the particles "leak before break" to use LWR terminology). Based on calculations performed for modular HTGRs, this failure mechanism is not an important contributor to fuel failure at normal operating temperatures. However, as discussed in Section 5.3.2, relatively high SiC degradation rates can occur if temperatures are higher than ~1800°C for extended periods of time (~100 hours or longer), and thermal decomposition of SiC occurs rapidly at temperatures above 2000°C. The maximum expected fuel temperatures in the Xe-100 during depressurized core heatup accidents are well below 2000°C as will be demonstrated through safety analyses in future licensing applications.

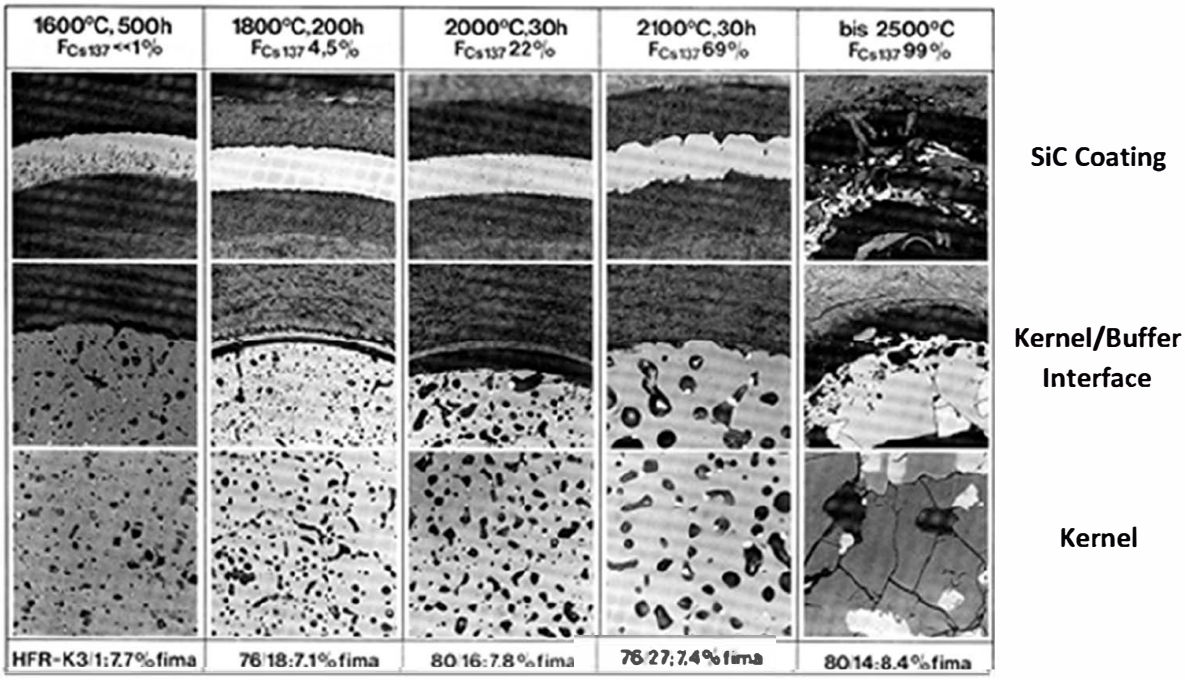

Figure 15: Thermal Decomposition of the SiC Coating

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

The fuel-service conditions and fuel-particle properties that influence the fuel-failure mechanisms are summarized in Table 4 [NGNP MST WP 2010] [55]. The fuel particles are designed and manufactured such that the properties defined in the table are within limits that result in acceptable fuel performance. The failure mechanisms are correlated with the reactor service conditions in models that are used to predict fuel performance. For the Xe-100, these are found in the integrated source term code suite XSTERM, as described further in X-energy's MST LTR [Loza 2021] [3].

Table 4. Relationship Between Fuel-Failure Mechanisms and Fuel-Particle Properties

	Parameters that strongly Influence the Failure Mechanism	
Failure Mechanism	**Service Conditions**	**Fuel-Particle Properties**
Pressure vessel failure	Temperature, burnup, fast fluence	Strength of SiC Buffer density (void volume) Fission-gas release Kernel type (CO production) Layer thicknesses IPyC and OPyC performance
Irradiation-induced IPyC failure and partial buffer/IPyC debonding leading to SiC cracking	Fast fluence, temperature	Degree of bonding between buffer and IPyC and IPyC and SiC Dimensional change of PyC Irradiation-induced creep of PyC Anisotropy of PyC Strength of PyC PyC thickness PyC density Tensile stress in SiC at IPyC crack tip SiC strength
IPyC partial debonding	Temperature, fast fluence	Nature of IPyC/SiC interface Interfacial strength Dimensional change of IPyC Irradiation-induced creep of IPyC
Heavy-metal dispersion in IPyC	Temperature, burnup	IPyC permeability
Kernel migration	Temperature, burnup, temperature gradient	Kernel type (UO_2, UCO, etc.) Buffer and IPyC thickness
Diffusive release through intact SiC layers	Temperature, burnup, temperature gradient, time at temperature	Chemical state/transport behavior of fission products Microstructure of SiC SiC thickness
Fission product attack of SiC	Temperature, burnup, temperature gradient, time at temperature	Chemical state/transport behavior of fission products Kernel type (UO_2, UCO, etc.) Microstructure of IPyC and SiC
Corrosion of SiC by CO	Temperature, burnup, time at temperature	Kernel type (UO_2, UCO, etc.) IPyC integrity
SiC thermal decomposition	Temperature, time at temperature	SiC thickness SiC microstructure
SiC permeability/SiC degradation	Burnup, temperature, fast fluence	Microstructure of SiC Thickness of SiC Permeability of SiC

5.1.5 TRISO Fuel Manufacturing and Quality Control

The processes for fabricating high-quality TRISO fuel are summarized in [Bresnick 1991 [68], TEC-1645 2010, [70] and TEC-1674 2012 [23]]; relevant subsections that describe those processes that will be used to fabricate LEU UCO TRISO-coated particles in spherical fuel elements for the Xe 100 are excerpted below. While these established fabrication protocols that have been developed internationally over several decades provide a sound technology baseline, it should be recognized that the AGR program continues to refine these fabrication processes in particular with the goal of process integration and optimization for the eventual mass production of TRISO fuel [e.g., Barnes 2008 [78], Phillips 2012 [79]].

5.1.5.1 UCO Kernel Fabrication

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]^{E,P}

Figure 16: [[ ]]^{P}

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]]^{E,P}

5.1.5.2 TRISO Coating

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]]^{E,P}

Figure 17: [[ ]][P]

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]]^{E,P}

5.1.5.3 Sphere Fabrication

The historic method for producing pebbles is described in this section. The spherical fuel element, as shown in Figure 3, consists of a central, spherical fuel zone of 50 mm diameter, in which the TRISO-coated particles are homogeneously distributed in a carbonaceous matrix material, and a 5 mm thick fuel-free shell of matrix material surrounding the fuel zone.

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Figure 18: [[ ]]^{P}

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]]^{E,P}

5.1.5.4 Fuel Compact Fabrication

The process used to fabricate cylindrical fuel compacts for prismatic fuel elements containing TRISO-coated particles is of interest here because the AGR program is qualifying LEU UCO TRISO-coated particles in cylindrical compacts. The pressures used to make fuel compacts are an order-of-magnitude lower than those used to make fuel spheres (typical compacting pressures are less than 20 MPa). Consequently, the Xe-100 fuel qualification program (Section 6) will need to demonstrate that these higher pressures for forming spheres have no deleterious effects on the UCO TRISO-coated particles relative to the results of the AGR program.

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5.1.5.5 Quality Control

Quality Control methods for the fabrication of TRISO fuel particles are well established and have been used for large-scale fuel production at GA in the U.S. and at NUKEM in Germany. [Bresnick 1991] [68] provides an overview of the fuel particle sampling techniques and QC inspection/testing methods used at GA for FSV fuel manufacturing and for fabrication of irradiation test fuel for the MHTGR and GT-MHR. The AGR program used these FSV methods as a point of departure and refined a number of them by taking advantage of modern analytical instrumentation with improved detection limits and accuracies (Section 5.3.2.1).

Given that the quantities of fuel particles in a modular HTGR reactor core number in the few billions, acceptance testing of the fuel particles and fuel spheres to determine conformance with specification requirements is necessarily performed on a statistical basis (i.e., statistical quality control). The statistical QC protocols adopted for the fabrication of TRISO fuel particles are summarized in Appendix B.

Additional QC methods have been developed as part of the German HTR program to characterize and control the attributes of fuel spheres containing TRISO-coated particles [e.g., TEC-1645 2010 [70] and TEC-1674 2012 [23]]. These German QC methods for spheres have also been used by the Chinese in manufacturing fuel spheres for HTR-10 [Zhao 2006] [61] and for HTR-PM [Zhou 2013] [10].

5.2 TRISO-COATED PARTICLE FUEL PERFORMANCE IN SPHERICAL FUEL ELEMENTS

Most of the high-quality TRISO-coated fuel particles in spherical fuel elements that have been produced to date have had LEU UO_2 kernels. There has been some experience in Germany with performance of HEU UCO TRISO fuel particles in spheres. Of the LEU UO_2 experience base, the most extensive contribution is from the earlier German program as summarized in Section 5.2.1. High-quality LEU UO_2 TRISO in spheres

was also produced by the Russians and the South African PBMR project on a laboratory scale, as summarized in Section 5.2.2 [NGNP FQ WP 2010] [4]. In addition, the Chinese are now mass producing high-quality LEU UO_2 TRISO fuel for their HTR-10 prototype reactor and for their HTR-PM commercial modular HTGRs as summarized in Section 5.2.3. The experience with HEU UCO particle fuel is spheres is described briefly in Section 5.2.4.

The experience described in this section will be applied by X-energy to the development of methods for fabrication and testing of TRISO-coated UCO fuel particles in spheres.

5.2.1 German LEU UO_2 TRISO Experience

The German LEU UO_2 TRISO fuel design evolved from decades of coated-particle fuel fabrication, irradiation, PIE, and safety testing experience covering a wide range of particle designs, fuel forms, and irradiation and testing conditions. An extensive German fuel-development program was conducted through the mid-1990s, resulting in a substantial body of fabrication, irradiation, and PIE and testing data [e.g., [NGNP FQ WP 2010] [4] [TEC-978 1997] [69], [TEC-1645 2010] [70], and [TEC-1674 2012] [23]].

5.2.1.1 Irradiation Testing

The German fuel irradiation experience includes both large-scale fuel testing in the AVR and carefully controlled irradiations in MTRs in Germany, the Netherlands, and France. The numerous tests of high-quality LEU 500-µm LEU UO_2 TRISO fuel are listed in Table 5.

Table 5. Normal Operation Prototype Sphere Irradiation Data Summary

ID	No. of Particles	Burnup (% FIMA)	Fast Fluence (10^{25} n/m$^{2)}$	Irradiation Temperature °C	PIH Temperature °C[10]	Exposed Kernels	SiC Defects
AVR Spheres (14 GLE-3 Spheres)							
AVR 88/15	16,400	8.7	2.4		1,600	0	0
AVR 82/20	16,400	8.6	2.4		1,600	0	1
AVR 88/33	16,400	8.5	2.3		1,600	0	3
AVR 71/22	16,400	3.5	0.5		1,600	0	0
AVR 82/9	16,400	8.9	2.5		1,600	0	N/D*
AVR 90/20	16,400	9.8	2.9		1,620 T	0	0

[10] "T" indicates that the fuel particles were subjected to a transient-simulation heat-up test after irradiation.

ID	No. of Particles	Burnup (% FIMA)	Fast Fluence (10^{25} n/m^{2})	Irradiation Temperature °C	PIH Temperature °C[10]	Exposed Kernels	SiC Defects
AVR 90/2	16,400	9.2	2.7		1,620 T	0	1
AVR 90/5	16,400	9.2	2.7		1,620 T	0	0
AVR 85/18	16,400	9.15	2.6		1,620 T	0	0
AVR 89/13	16,400	9.1	2.6		1,620 T	0	0
AVR 74/11	16,400	6.2	1.4		1,700	0	1
AVR 91/31	16,400	9.0	2.6		1,700 T	0	0
AVR 88/41	16,400	7.6	1.9		1,800	0	2
AVR 76/18	16,400	7.1	1.7		1,800	0	0
Materials Test Reactor Spheres (8 Phase 1 LEU + 8 Proof Test)							
HFR-K3/1	16,400	7.5	4	1,200	1,600	0	0
FRJ2-K13/2	16,400	8	0.2	1,150	1,600	0	1
FRJ2-K13/4	16,400	7.6	0.2	1,120	1,600	0	0
HFR-K3/3	16,400	10.6	5.9	920	1,800	0	0
HFR-K3/2	16,400	10	5.8	920		0	N/D
HFR-K3/4	16,400	9	4.9	1,220		0	N/D
HFR-K5/1	14,580	7.8	4	923		0	N/D
HFR-K5/2	14,580	10.1	5.8	909		0	N/D
HFR-K5/3	14,580	10.3	5.9	903		0	N/D
HFR-K5/4	14,580	9.3	4.9	921		1	N/D
HFR-K6/1	14,580	8.3	3.2	1,090		0	N/D
HFR-K6/2	14,580	10.6	4.6	1,130		0	N/D
HFR-K6/3	14,580	10.9	4.8	1,140		0	N/D
HFR-K6/4	14,580	9.9	4.5	1,130		2	N/D

ID	No. of Particles	Burnup (% FIMA)	Fast Fluence (10^{25} n/m^{2})	Irradiation Temperature °C	PIH Temperature °C[10]	Exposed Kernels	SiC Defects
FRJ2-K13/1	16,400	7.5	0.2	1,125		0	N/D
FRJ2-K13/3	16,400	7.9	0.2	1,150		0	N/D

Analysis Summary of Irradiated Spheres

Parameter	Number of Particles	Total Particles	Maximum Parent Population Particle Fraction	
Confidence Level that Indicated Particle Fraction is not Exceeded in Parent Population			50%	95%
Exposed Kernels	3	477,400	7.6×10^{-6}	1.6×10^{-5}
SiC Defects	9	278,800	3.5×10^{-5}	5.6×10^{-5}

* N/D – Not Determined.

These data and the subsequent analyses lead to the following observations and conclusions regarding LEU UO_2 TRISO behavior during normal operation:

- Exposed kernels:
  - In-pile fission gas release data from eight LEU Phase 1 and eight Proof Test spheres containing a total of 247,800 particles irradiated in MTRs indicated three particles with exposed kernels at the beginning of irradiation, with no additional failures during irradiation.
  - Post-irradiation heating data from 14 GLE-3 spheres containing 229,600 particles irradiated in the AVR indicated no particles with exposed kernels based upon low release rates of ^{85}Kr.
- SiC defects and failures:
  - The nominal as-manufactured free-uranium fraction for GLE-3 spheres irradiated in the AVR was 5.07×10^{-5}, and for LEU Phase 1 spheres irradiated in MTRs, it was 3.5×10^{-5}. These results include both exposed kernels and SiC defects (as determined by burn/leach measurements).
  - Post-irradiation heating cesium release data from 13 GLE-3 spheres containing 213,200 particles irradiated in the AVR and four LEU Phase 1 spheres containing 65,600 particles irradiated in MTRs (a total of 247,800 particles) indicated nine particles with defective or failed SiC layers. These data result in a 50% confidence SiC defect/failure fraction of 3.5×10^{-5} and a 95% confidence value of 5.6×10^{-5}.
- The weighted as-manufactured SiC defect fraction for the 13 GLE-3 spheres and four LEU Phase 1 spheres, discussed above, is 4.9×10^{-5}. This is more than the 50% confidence post-irradiation SiC

defect/failure fraction (3.5×10^{-5}) and is less than the 95% confidence value (5.6×10^{-5}), indicating that no statistically significant SiC failures occurred during irradiation and that the as-manufactured burn-leach results were dominated by SiC defects.

- Both the in-pile gas release (exposed kernels) and the post-irradiation heating cesium release (SiC defects) indicated no particle failures during irradiation in a total of 477,400 particles. These results indicated that particle failures during irradiation are highly unlikely.

Two additional irradiations of archived German fuel spheres, designated HFR-EU1 and HFR EU1bis, were conducted [TEC-1674 2012 [23]]. The HFR-EU1bis irradiation of five German AVR 21-2 fuel spheres began in September 2004 and was completed in October 2005. The intended irradiation plan was to maintain the central temperature of all spheres at approximately 1,250°C. Early in the irradiation, an operating error resulted in inadvertent introduction of pure neon, resulting in temperatures well above the target values. Post-irradiation thermal modeling of operation with pure neon indicated a temperature at the outer graphite shroud radius of 1,350°C, which could result in sphere centerline temperatures approaching 1,600°C for an extended period (These temperatures are well above those at which the Xe-100 fuel will operate.) Gaseous fission product release data indicate that one or more exposed kernels were present at the beginning of the irradiation, and several more particles failed during the irradiation. The end-of-irradiation burnup ranged from 9.3 to 11% FIMA.

The HFR-EU1 experiment contained three spheres of archived German fuel from the AVR 21-2 batch and thus had essentially the same properties as those irradiated in HFR-EU1bis (the test also included two Chinese fuel spheres fabricated by INET). The irradiation was completed in two campaigns. Initial operation began in September 2006, and the test was removed from the reactor in February 2008 because of operational concerns associated with the number of thermocouple failures experienced. Irradiation was resumed in October 2009 and completed in February 2010. The German capsule was controlled to achieve a sphere surface temperature of approximately 900°C, and a maximum burnup of 11.2% was achieved. Gaseous fission-product release data indicate that no exposed kernels were present at the beginning of the irradiation, and no failures occurred during the irradiation.

5.2.1.2 Safety Testing

A large number of post-irradiation heating tests were performed with German LEU UO_2 fuel spheres irradiated in the AVR or MTRs (Table 6). The results of the German heating tests are summarized in Table 6 and Table 7. Table 6 gives the exposed kernel fraction as inferred from the measured ^{85}Kr fractional release, and Table 7 gives the SiC failure fraction as inferred from the measured ^{137}Cs fractional release.

Table 6. Summary of Heating Test Krypton Release Results

Parameter	Test Temperature			
	1,600°C	1,700°C	1,800°C	1,800°C[a]
Average Burnup, % FIMA	8.3	7.6	6.5	8.4
Average Fast Fluence, 10^{21} n/cm^2	2.2	2.0	1.8	2.6
Number Particles	213,200	36,132	83,631	50,831
Number Exposed Kernels	6	20	69	12

Parameter	Test Temperature			
	1,600°C	1,700°C	1,800°C	1,800°C[a]
Exposed Kernel Fraction[b] (50% confidence)	3.1×10^{-5}	5.72×10^{-4}	8.33×10^{-4}	2.49×10^{-4}
Exposed Kernel Fraction[b] (95% confidence)	5.6×10^{-5}	8.04×10^{-4}	1.01×10^{-3}	3.82×10^{-4}

[a]Excludes AVR 74/10 and 70/33 test data.
[b]Maximum parent-population exposed-kernel fraction.

Table 7. End-of-Irradiation Fuel Condition Inferred from Heating Test Data

ID	Number of Particles	Burnup (% FIMA)	Fast Fluence (10^{25} n/m^2)	Irradiation Temperature (°C)	Test Temperature (°C)	Exposed Kernels	SiC Defects
AVR Spheres							
AVR 88/15	16,400	8.7	2.4		1,600	0	0
AVR 82/20	16,400	8.6	2.4		1,600	0	1
AVR 88/33	16,400	8.5	2.3		1,600	0	3
AVR 71/22	16,400	3.5	0.5		1,600	0	0
AVR 90/20	16,400	9.8	2.9		1,620 T	0	0
AVR 90/2	16,400	9.2	2.7		1,620 T	0	1
AVR 90/5	16,400	9.2	2.7		1,620 T	0	0
AVR 85/18	16,400	9.15	2.6		1,620 T	0	0
AVR 89/13	16,400	9.1	2.6		1,620 T	0	0
AVR 74/11	16,400	6.2	1.4		1,700	0	1
AVR 91/31	16,400	9.0	2.6		1,700 T	0	0
AVR 88/41	16,400	7.6	1.9		1,800	0	2
AVR 76/18	16,400	7.1	1.7		1,800	0	0
MTR Spheres							
HFR-K3/1	16,400	7.5	4	1,200	1,600	0	0
FRJ2-K13/2	16,400	8	0.2	1,150	1,600	0	1
FRJ2-K13/4	16,400	7.6	0.2	1,120	1,600	0	0
HFR-K3/3	16,400	10.6	5.9	920	1,800	0	0
Total Particles	278,800	8.2	2.3	←Average Values Totals→		0	9

ID	Number of Particles	Burnup (% FIMA)	Fast Fluence (10^{25} n/m^2)	Irradiation Temperature (°C)	Test Temperature (°C)	Exposed Kernels	SiC Defects
50% confidence maximum parent population fraction						2.49×10^{-6}	3.47×10^{-5}
95% confidence maximum parent population fraction						1.07×10^{-5}	5.63×10^{-5}

Representative examples of the German PIH tests of LEU UO_2 TRISO fuel are given below. The ^{85}Kr and ^{137}Cs release from LEU UO_2 TRISO fuel during 1600°C isothermal heating tests are shown in Figure 19 [Hantke 1992] [83]. The initial ^{85}Kr responses indicate that there were no through-coating particle failures at the end of irradiation for any of the spheres. The ^{137}Cs responses indicate one particle with a SiC defect in sphere FRJ2 K13/2, and no particles with SiC defects in the other two spheres at the end of irradiation. The beginning of an upward trend in the cesium release from HFR K3/1 after 200 hr may indicate the onset of increasing permeability in a SiC layer, with the level of release remaining an order of magnitude below a single-particle inventory after 300 hours.

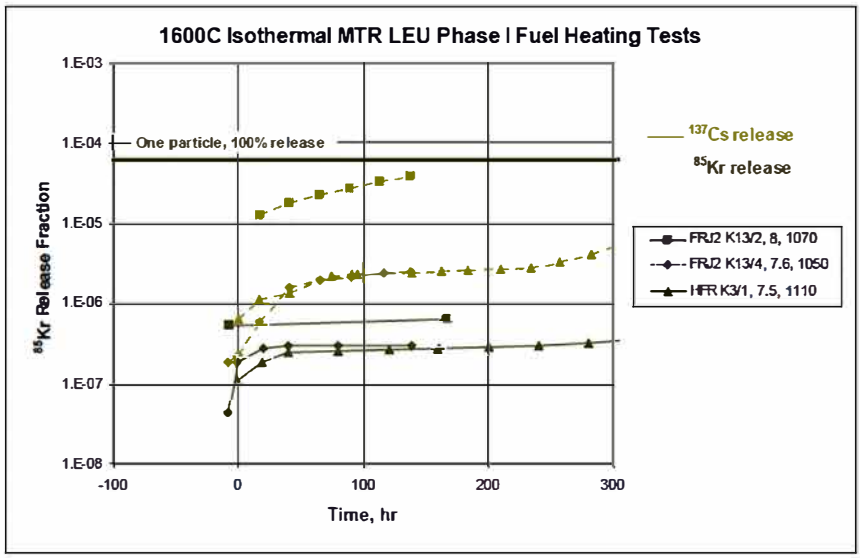

Figure 19: Release of ^{85}Kr and ^{137}Cs from German LEU UO_2 TRISO Fuel at 1600°C

The ^{85}Kr and ^{137}Cs releases from LEU UO_2 TRISO fuel during 1800°C isothermal heating tests are shown in Figure 20 [Hantke 1992] [83]. The initial ^{85}Kr responses indicate there were no through-coating particle failures at the end of irradiation. The relatively rapid increase in ^{137}Cs release from two spheres indicates degradation of the SiC coating, probably a result of CO corrosion. After ~50 hours the increase of ^{85}Kr release indicates through-coating failure of at least one particle in the FRJ2 K13/4 sphere and multiple through-coating failures in the HFR K3/3 sphere.

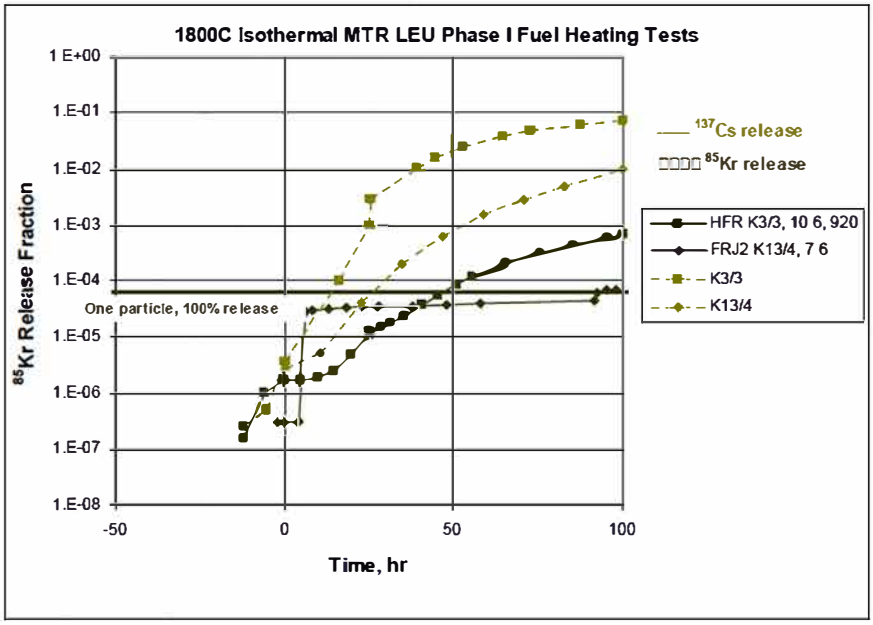

Figure 20: Release of ^{85}Kr and ^{137}Cs from German LEU UO_2 TRISO Fuel at 1800°C

The release of fission metals from an AVR sphere with LEU UO_2 TRISO fuel during a 1800°C isothermal heating test is shown in Figure 21 [Hantke 1992] [83]. Again, there is clear evidence from the ^{137}Cs and ^{90}Sr releases of SiC degradation in a significant number of particles with the apparent SiC failure fraction approaching 1% after 100 hr.

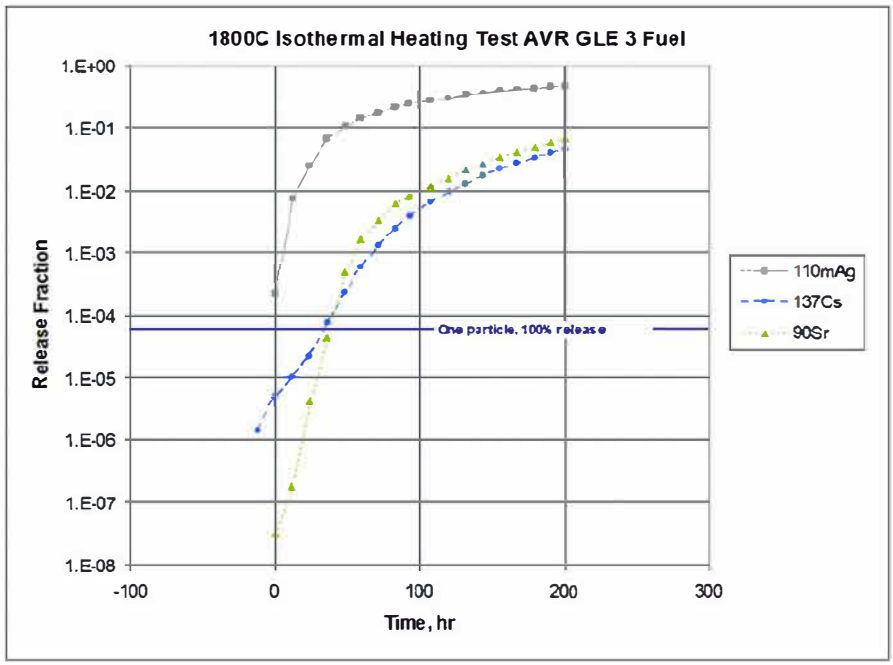

Figure 21: Fission Metal Release from an AVR Sphere with LEU UO_2 Fuel at 1800°C

These data support four important overall conclusions with regard to the performance of LEU UO_2 TRISO-coated particles during core heatup accidents:

- Substantial margins exist in the ultimate capability of the fuel relative to typical pebble-bed design requirements. Only a small fraction of the fuel spheres in the pebble-bed core would experience the maximum temperature of <1600°C and only for a short time relative to the hold times in the PIH tests.
- No exposed-kernel failures were observed in any of the MTR irradiations.
- The 50% confidence SiC-defect fraction after irradiation to substantial burnups in both the AVR and MTRs and subsequent heating at 1,600°C is approximately equal to the mean free uranium fraction of the as-manufactured fuel, indicating no additional SiC defects were produced during irradiation.
- The degradation in fuel performance at elevated temperatures is regular and gradual. No sudden changes in behavior ("cliff-edge" effects) as a function of irradiation temperature, burnup, or accident temperature were observed. However, significant SiC degradation does occur in UO_2 particles at 1800°C within ~50 hours, probably as a result of CO corrosion.

While these data are for LEU UO_2 TRISO-coated particles, they are generally expected to apply equally well to LEU UCO TRISO data. In fact, the AGR data indicate that the margins are even larger with UCO TRISO-coated particles (Section 5.3.2)

The data base produced by the German heatup tests includes fuels that were irradiated in both the AVR reactor (i.e., prototypical temperature-varying environment of the pebble-bed core) and in MTR isothermal tests. The data are generally consistent and coherent with regard to irradiation temperature, burnup, fluence, and heating test temperature, notwithstanding the statistical limitations of dealing with very low probabilities of particle failures, the random nature of outlier particle defects (e.g., SiC layer flaws, faceting) and characteristics (e.g., combinations of kernel diameter and layer thicknesses), and the potential for failure modes with varying ^{85}Kr release characteristics. The overall consistency of these data should adequately address the concern raised by the NRC staff [NRC 2014] [43] about potential effects of neutron spectral differences between water-cooled MTRs and an actual operating modular HTGR.

5.2.2 PBMR LEU UO_2 Experience

Pebble Bed Modular Reactor (Pty) Ltd (PBMR) was a South African-based company with international ownership, including Westinghouse and BNFL. PBMR was created to design and build pebble-bed modular reactors that were based upon the German pebble-bed reactor design and fuel technology. PBMR was an active participant in the NGNP Project, wherein it was represented by Westinghouse Electric, one of the corporate owners. The NGNP FQ and MST white papers were written to address both U.S. prismatic and PBMR pebble-bed designs. A PBMR fuel qualification white paper was drafted by Westinghouse, and much of its content was incorporated into [NGNP FQ WP 2010] [4] Because of a change in South African government policy, the PBMR Project was suspended in 2010, and the NGNP/NRC interactions related to the pebble-bed option were tabled, including the generation of responses to the NRC requests for additional information (RAIs) on pebble-fuel qualification and performance described in NGNP FQ and MST white papers.

The reference fuel design for the PBMR Project was essentially the same as the LEU UO_2 TRISO fuel design for the Siemens HTR-Modul. The PBMR Project had planned and initiated a comprehensive FQ program which was described in the FQ white paper for the NGNP Project. The PBMR fuel qualification strategy was to build upon the considerable success of the German pebble fuel development program (Section

5.2.1), but it also included supplemental testing to demonstrate to the regulatory agencies that their reference fuel performed at least as well as the German fuel. Consequently, the PBMR-based sections of the NGNP FQ white paper contained an extensive discussion of the German LEU UO_2 data base [NGNP FQ WP 2010] [4].

The planned PBMR supplemental program to augment the extensive German LEU UO_2 data base consisted of the following elements:

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

This supplemental program included essentially the same standard elements as the AGR fuel program [AGR TDP 2008] [84]: (1) fuel process optimization, including characterization of the as-manufactured fuel; (2) irradiation testing in MTRs; (3) PIE and (4) post-irradiation heating tests. The planned irradiation tests are summarized in Table 8.

Of special note is the irradiation of LEU UO_2 TRISO-coated particles fabricated by PMBR and compacted by ORNL in the AGR-2 test (one of six capsules in the test train was dedicated to PBMR fuel; see Section 5.3.2.2.1. The AGR-2 irradiation has been completed, and a limited amount of PIE and safety testing of the PBMR fuel is reportedly planned. However, all of the test results for the PBMR fuel, including the on-line release rate-to-birth rate ratio (R/B) data, is classified as proprietary Information; hence, none is available for inclusion herein at this writing.

Table 8. Planned PBMR Fuel Qualification Program

Test	Fuel to be Tested		Test Description						Technical Objective
	Production Source	No.	Test Reactor	Burnup (% FIMA)	Temperature (°C)	Cycle	Post Heat	PIE	
AGR-2	FDL[a]	Particles in Compacts	ATR	> 10	1,150	No	No	Y	Performance demonstration of irradiated particles
Pre-qualification fuel testing	FDL	5 spheres	HFR[b]	10.1±1	1,200 / 1,100	No	No	Y	Confirm the capability of the FDL fuel
Pre-qualification fuel testing	FDL	4 spheres	IVV-2M[c]	10.1±1	1,200 / 1,100	No	No	Y	Confirm the capability of the FDL fuel
Partial Burnup Demonstration	Pebble Fuel Plant	4 spheres	IVV-2M	{5}	1,200	No	Four fuel spheres to 1,600°C, then to 1,800 °C	Y	Qualification to 5% FIMA. Zero or low number of coated particle failures.
Full Burnup Demonstration	Pebble Fuel Plant	12 spheres	IVV-2M	{~9.8}	900 / 1,150	Yes	Eleven fuel spheres, five to 1,600°C, six to 1,800°C	Y	Full fuel proof test, including simulated PLOFC transient in test reactor. Detailed PIE.
Matrix Graphite testing	Pebble Fuel Plant	196[11]	IVV-2M	-	900 / 1,100	No	No	Y	Confirm matrix graphite irradiated properties

[a]FDL: fuel development laboratory (i.e., in general, lab-scale equipment but with a commercial-size coater).
[b]HFR: High Flux Reactor, Petten, NL (extensively used to test German UO_2 fuel) [e.g., TEC-1674 2012] [23]
[c]IVV-2M: light-water MTR, Zarechny, RF (previously used to irradiate Chinese fuel spheres) [Koscheev 1999] [85]

5.2.3 Chinese LEU UO_2 TRISO Experience

The TRISO-coated particle fuel development program in China was initially established to support the construction and operation of the 10 MWt HTR-10 pebble bed reactor [Wu 2002] [86]. Development of fuel fabrication methods was based on the German LEU UO_2 TRISO-coated particle and spherical fuel element design, using fuel fabrication equipment that was purchased from the German particle-fuel manufacturer NUKEM. Fabrication of fuel for the first core of HTR-10 began in December 1999 with

production of 11,700 fuel spheres by September 2000, a sufficient number to support initial criticality [Zhao 2006] [61]. The low power level, combined with the replication of the German LEU UO_2 fuel design, which enabled the use of the German fuel-performance data base, facilitated the demonstration of large margins to fuel service limits. Thus, the fuel irradiation and testing program was conducted in parallel with initial operation of HTR-10 [Zhong 2001] [87].

The fuel quality, as indicated by the heavy-metal contamination fraction in the fuel spheres, improved by more than an order of magnitude during the course of fuel production for the HTR-10 core (a total of 25 batches of spheres). Heavy-metal contamination (as measured by the leach-burn-leach procedure) in the early batches was typically ~10^{-4}, while in the last fifteen batches it was typically ~10^{-5} and lower [Zhao 2006] [61].

Irradiation of four fuel spheres taken from early in the first HTR-10 core production began in the Russian IVV-2M reactor in July 2000 and was completed in February 2003 [Tang 2004] [88]. The irradiation rig contained five capsules; capsules 2 through 5 contained fuel spheres, and capsule 1 contained matrix graphite specimens. The irradiations were conducted at ~1,000°C, with short-term increases to ~1,200°C, and to burnups ranging from 95 to 107 GWd/MtU. In-pile gas release measurements indicated the presence of one or two exposed kernels in two of the irradiated spheres from the beginning, consistent with the as-manufactured free-uranium measurements for early production batches. One of the capsules failed during irradiation with the attendant loss of R/B data, and PIE showed substantial damage to the sphere (in-pile gas-release measurement capability failed when the capsule failed). Another capsule experienced high temperatures at the end of the irradiation when a control thermocouple failed, resulting in temperatures well beyond the planned conditions, and a significant number of particles failed as a result. On-line R/B data indicated that no failures had occurred earlier during irradiation when conditions remained within specified limits.

An additional irradiation of fuel spheres produced at the Institute of Nuclear and New Energy Technology (INET) for HTR-10 was conducted in the HFR Petten reactor in an experiment designated HFR-EU1 [Marmier 2008 [89] and TEC-1674 2012 [23]]. The HFR-EU1 experiment included two spheres from China and three from the German program (from AVR reload 21-2) with the two fuels placed in separate capsules. A primary objective of the irradiation was to subject the spheres to high burnups (17% FIMA for Chinese spheres, 20% FIMA for German spheres) to investigate the ultimate capability of the coated-particle design developed in Germany. The irradiation was interrupted after reaching burnups of 8% FIMA (Chinese spheres) and 11% FIMA (German spheres) by a high level of thermocouple failures resulting in a large temperature excursion. The irradiation was subsequently completed, and a significant amount of particle failure as a result of the test excursion was confirmed during the PIE [TEC-1674 2012] [23].

With the successful operation of HTR-10, the Chinese moved to commercialize modular HTGR technology by initiating the HTR-PM Project [Zhang 2016] [90]. Two 250-MWt pebble-bed modules are under construction at the Shidao Bay Nuclear Power Plant in Shandong Province, and together these two reactor modules will drive a single steam turbine generating 200 MWe. The first concrete was poured in December 2012 and the reactor modules have each completed cold commissioning as of late 2020.

The fuel design for HTR-PM is essentially the same as the LEU UO_2 TRISO fuel used for HTR-10 except that the nominal thickness of the fuel-free zone is reduced to 4 mm from the 5 mm thickness used internationally for all previous pebble fuel designs [Zhou 2013] [10]. To facilitate mass production of fuel

elements, the manufacturing process has been optimized. A new particle overcoating system has been developed with an improved efficiency, and an optimized carbonization process promises to reduce the carbonization time from >70 hr to ~20 hr. Properties of the manufactured fuel spheres and matrix graphite spheres have met the design specifications for HTR-PM.

The Chinese have constructed and brought on-line last year a 300,000 sphere/year fuel factory in Baotou, Inner Mongolia, to service the two HTR-PM modules [WNA 2016] [91]. The factory is based on a trial production line developed by INET at Tsinghua University to produce 100,000 spherical fuel elements/year. Fuel production at Baotou started in March 2016.

A so-called "proof test" of the HTR-PM fuel has been performed in HFR Petten, although the pebbles did not come from the 300,000 sphere/year Baotou facility itself. [Knol 2016] [92]. Instead, the pebbles were made on equipment identical to that at the INET facility and replicated to achieve commercial scale at Baotou. Five spheres containing 12,000, 17%-enriched, 500-µm LEU UO_2 TRISO-coated particles were irradiated at a nominal sphere centerline temperature of 1050 ± 50°C to a peak burnup of 112 MWd/kgHM. The reported R/B measurements were remarkably low as shown in Figure 22. These reported R/B results are substantially lower than those obtained in other fuel sphere irradiations, and the reasons for this result are not currently understood.

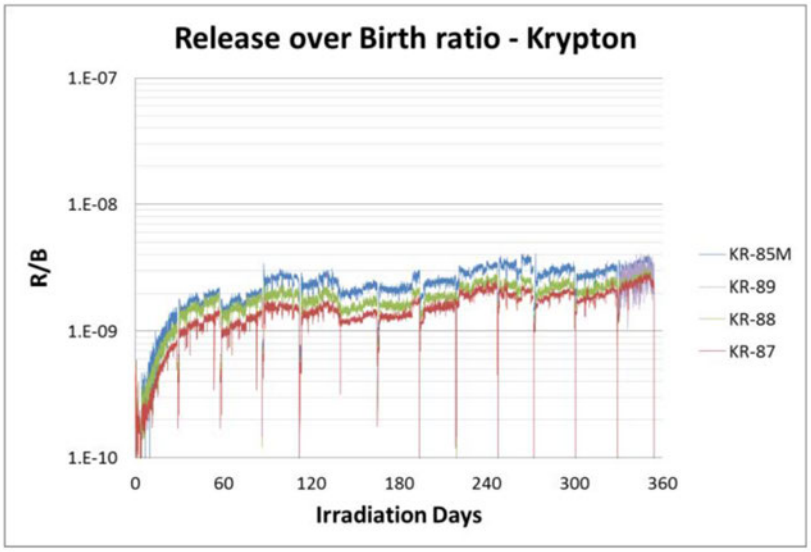

Figure 22: Measured Kr R/Bs for HTR-PM Proof Test Fuel in HFR Petten

After a non-destructive PIE at HFR Petten, the spheres have been shipped to the Joint Research Centre – Karlsruhe, Germany, for PIH tests in the KÜFA furnace. Based upon the measured R/B data, this has been a highly successful proof test to this point. However, although this Petten test is reported as a proof test of HTR-PM fuel, the test fuel was actually manufactured at INET rather than at the new fuel factory in Baotou, which will produce the reactor fuel. In 2013 construction began on an HTGR fuel-production factory in Baotou, Inner Mongolia. Commissioning and trial production began at the plant in 2015, and in July 2017, China began mass production of the fuel. The production line, with capacity to produce 300,000 spherical fuel elements annually, has already (Sept. 2020) delivered the 200,000th spherical fuel element, marking the transition from test production to industrial production. Whether the fuel produced at Baotou will also be proof tested is unknown at this writing.

5.2.4 German HEU UCO TRISO Experience in Spheres

In 1977, 5,354 fuel spheres (about 21% of the full AVR core) containing HEU UCO fuel kernels were inserted into the AVR. This was the first large-scale test of UCO TRISO fuel in Germany. The fission-gas release from the AVR core remained low (2-3 × 10^{-5} R/B Kr-85m) while the UCO fuel spheres were under irradiation. However, HEU fuel development was discontinued in Germany because of the international non-proliferation treaty signed in 1977. In 1982, the German HTGR program selected LEU UO_2 TRISO for its reference fuel; consequently, no significant PIE or safety tests were performed on these HEU UCO fuel spheres irradiated in the AVR. The demonstrated ability to mass produce high quality HEU UCO TRISO fuel spheres implies that it should be possible to mass produce the same fuel type using LEU in place of HEU.

5.3 TRISO-COATED PARTICLE UCO FUEL PERFORMANCE IN COMPACTS

This section describes the successful experience to date in fabrication and testing of TRISO-coated UCO fuel particles in cylindrical fuel compacts for prismatic fuel modular HTGRs. The experience described in Section 5.2 with fabrication and testing of TRISO UO_2 in spheres and that described in this section for TRISO-coated UCO fuel particles will be combined by X-energy to develop processes for fabrication of TRISO-coated UCO fuel particles in spheres and successful testing of that fuel for the Xe-100

5.3.1 Historical U.S. Experience with TRISO Fuel in Compacts

The fuel element that forms the basis for U.S. prismatic modular HTGR designs is shown in Figure 23 [e.g., CDR 2010] [58]. The first prismatic fuel elements, which were developed and utilized in the FSV HTGR, utilized uranium/thorium carbide fissile and thorium carbide fertile particles pressed into cylindrical compacts and loaded into fuel holes drilled in prismatic-shaped graphite blocks [McEachern 2001] [15]. The NGNP prismatic design utilized TRISO particles with UCO kernels in compacts [NGNP FQ WP 2010] [4]. Experience with fabrication, irradiation, safety testing, and analysis methods for fuel to be used in prismatic fuel elements is discussed briefly in this section with the emphasis on experience with UCO TRISO fuel.

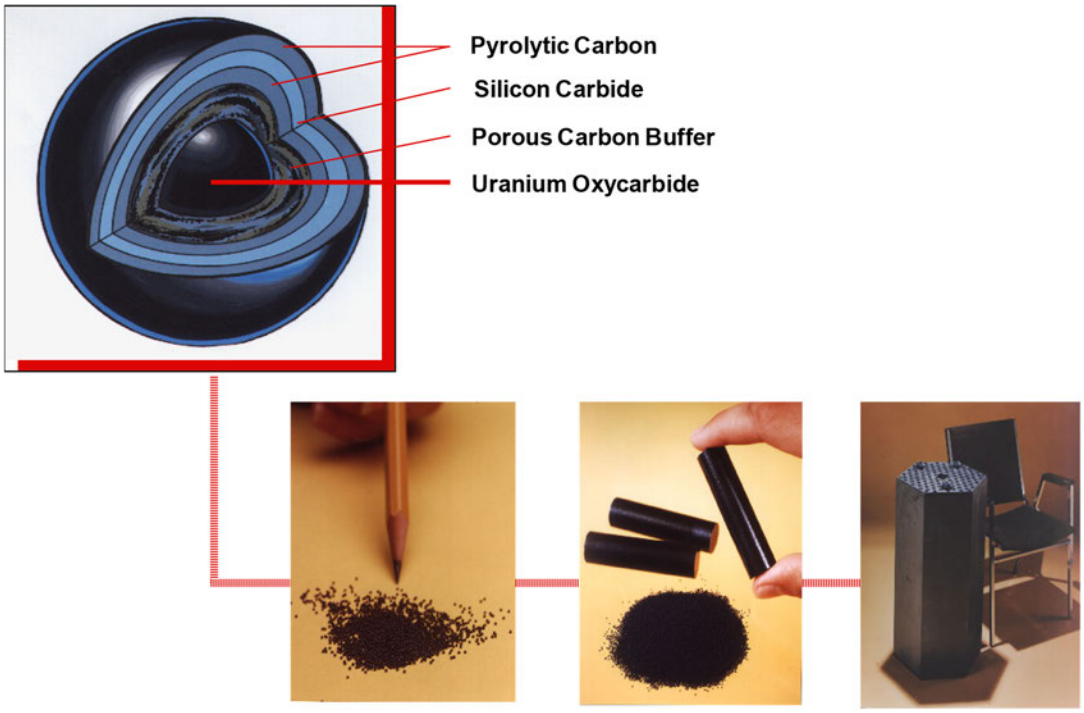

Figure 23: U.S. Prismatic Fuel Element

5.3.1.1 Irradiation of Early UCO TRISO-Coated Particles

The motivation for the development of UCO TRISO fuel, which began in U.S. in the late 1970s, was its promise of superior performance compared to other kernel compositions [NGNP FQ WP 2010] [4]. Despite its promise, which was based primarily upon on thermochemical analysis (Section 5.1.3), the UCO fuel in U.S. irradiation tests prior to the AGR program did not meet the stringent performance requirements for modular HTGR designs. These early UCO irradiations included R2 K13 [Myers 1985] [93], HRB-15A [Ketterer 1984] [94], HRB-15B [Ketterer 1981] [95] and HRB-16 [Ketterer 1985] [77]. However, the reasons for this unexpected performance were determined to be unrelated to the performance of the UCO kernel. Rather, the unexpected performance resulted from either as-manufactured defective SiC coatings, which were created during coating and/or compacting processes, or from a TRISO coating design that was inferior compared to the successful German TRISO coating design [Petti 2002] [82].

The most extreme example of an inferior coating design was the TRISO-P particle [Leikind 1993] [74]. Capsule HRB-21 and three New Production Reactor (NPR) capsules all contained TRISO-P UCO particles. The TRISO-P design featured both a significantly thicker and denser IPyC layer and an added porous “protective” (P-PyC) outer layer. Both design changes were made to solve perceived problems during fuel fabrication. The IPyC layer was thickened and made less porous (and more anisotropic) to improve the quality of SiC coating by reducing the potential for heavy-metal dispersion. The outer P-PyC layer was added to reduce the potential for introducing SiC defects from particle-to-particle contact during compacting. The design changes resolved these process issues, and the as-manufactured quality of the fuel compacts was dramatically improved. However, under irradiation the thicker, more anisotropic IPyC developed radial cracks which served as stress risers in the SiC layer, and the porous P-PyC layers shrank excessively which caused a high fraction of the OPyC layers to fail [Leikind 1993 [74] and Hobbins 1993

[75]]. The combined result of these design "improvements" was an order-of-magnitude increase in the in-service failure rates compared to that of conventional U.S. TRISO-coated particles even though the as-manufactured quality had been much improved. The TRISO-P coating design was abandoned based on this experience.

The German capsule FRJ2-P24 test, an irradiation of UCO TRISO-coated particles in cylindrical compacts under representative prismatic HTGR temperatures and burnup (but with very low fast fluence), showed excellent fuel performance with respect to on-line R/B measurements. German-made, 300-µm, 20% enriched UCO TRISO-coated particles in cylindrical fuel compacts were irradiated in this capsule. The UCO fuel achieved a burnup of up to 22% FIMA at a time-averaged temperature of ~1,120°C with no in-service coating failures observed. No kernel migration or SiC corrosion because of FP attack was reported [Borchardt 1982 [96] and Bauer 1983 [97]]. No PIH tests were performed with this UCO fuel since the Germans had adopted LEU UO_2 TRISO as their reference fuel.

Given the relatively poor performance of early U.S. UCO TRISO fuel, especially the TRISO-P fuel, the DOE initiated the NGNP/AGR Fuel Program to develop and qualify fuel for the NGNP and follow-on modular HTGRs. As described in Section 5.3.2, the AGR UCO TRISO-coated particle has been remarkably successful.

5.3.1.2 Safety Testing of Early UCO TRISO-Coated Particles

There was a very limited amount of PIH testing of UCO TRISO fuel prior to the AGR program. Moreover, these PIH tests typically went to >2000°C such that ~100% particle failure eventually occurred.

In one test series, 186 initially intact LEU UCO fuel particles from HRB-15A and HRB-15B were heated in both temperature ramp and isothermal accident simulation tests [Goodin 1985] [98]. The temperature ramp tests covered the range from ~1100°C to as high as ~2700°C, with heating rates in the range of ~19°C/h to ~190°C/h. The isothermal heating tests were conducted at 2050°C, 2200°C, and 2400°C. These temperatures and heating rates were representative of those expected in the large (>2000 MWt), non-modular HTGR designs under consideration at that time. The test series involved heating of the following other TRISO fuel types:

- HEU, LEU, and depleted UC_2
- ThO_2
- $(Th,U)O_2$
- LEU UO_2 and UO_2*[12]
- ThC_2 and $(Th,U)C_2$

Figure 24 (taken from [Goodin 1985] [98]) summarizes the results of 30-hour ramp heating tests for the various LEU fuels and HEU UC_2 fuel. The primary mechanism for TRISO coating failure and the attendant Kr-85 release under the simulated large HTGR accident conditions was found to be thermal decomposition of the SiC layer followed by either diffusion of fission products through the PyC coatings or breakage of the PyC coatings. Within the temperature range tested, fuel particle performance was found to depend

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

on the inherent thermal stability of the SiC coating layer and not to be measurably dependent upon burnup, fast neutron fluence, or kernel composition.

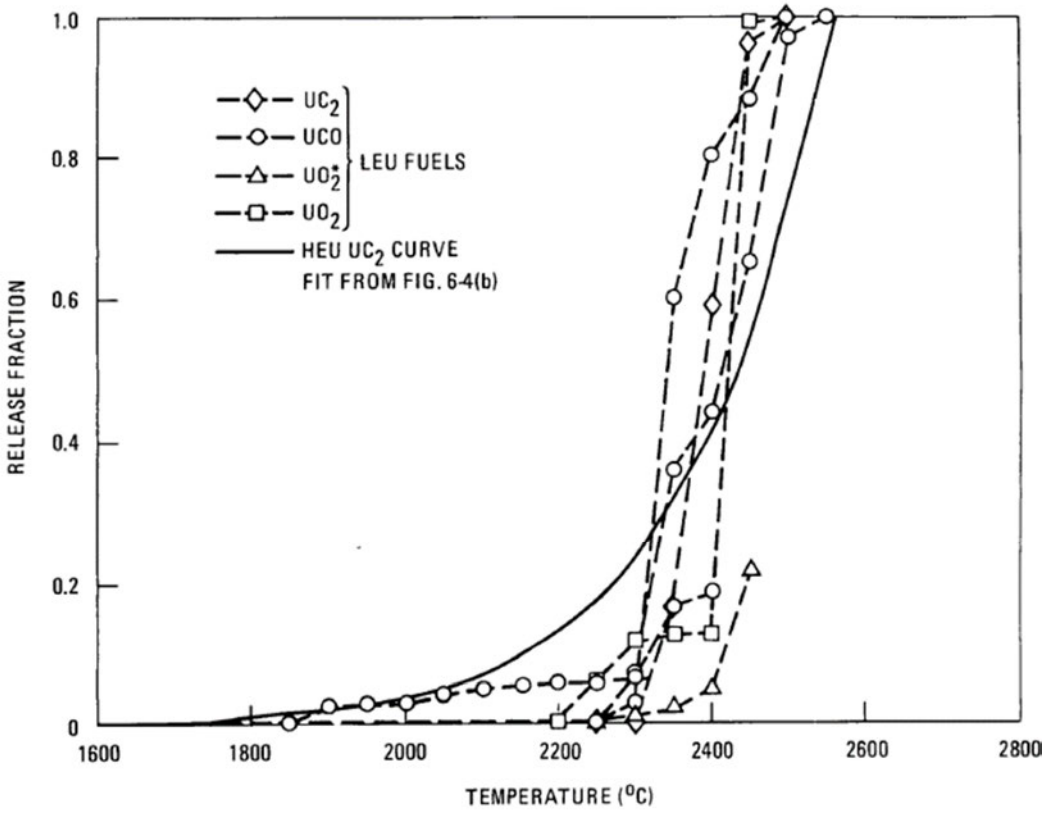

Figure 24: Kr-85 Release for Ramp Heating Tests of Candidate HTGR Fuel Types

Given the different chemical compositions of these fuel types, the similarity of the release profiles in Figure 24 is strong evidence that the performance of the fuel particles for the large HTGR accident conditions simulated in this heating test series is independent of kernel composition and depends only on the TRISO coating, specifically the thermal stability of the SiC coating. However, it should be noted that the temperatures associated with the large HTGR accident conditions are much higher than those during core heatup accidents in modular HTGR designs.

In another heating test series, 30 initially intact LEU UCO fuel particles irradiated in HRB-15B were heated isothermally for 10,000 hours at temperatures of 1200°C, 1350°C, or 1500°C (10 particles at each temperature) [Bullock 1984] [99]. LEU UO_2, UC_2, and two variations of UO_2* were also tested under the same conditions. With respect to the relative heating test performance of the UCO and UO_2 particles, the following differences were observed:

- At 1500°C, Eu-154 release started much earlier for the UCO fuel particles than the UO_2 particles, and the total Eu-154 release from the UCO particles (~50%) was higher than from the UO_2 particles (~15%). The UCO particles also released Eu-154 at both 1200°C and 1350°C, but the amount released decreased significantly with temperature. The UO_2 particles did not release Eu-154 at 1200°C or 1350°C.

- At 1500°C, Ag-110m release started much earlier in the UO_2 particles than the UCO particles, and the total Ag-110m release from the UO2 particles (~90%) was considerably higher than from the UCO particles (<10%).
- Cs-137 was released only at 1500°C and only from 3 of the 150 particles tested. Two of these were UO_2 particles. Diffusion through flawed, but nominally intact, SiC layers was apparently responsible for the steadily increasing release from the two UO_2 particles. None of the UCO particles released Cs-137 at any test temperature.

Although the sample sizes were exceedingly small (10 particles) and the coating conditions for depositing the SiC coatings on the UCO and UO_2 particles were not optimal, these initial conclusions about the differences between the fission release behavior of UCO and UO_2 TRISO-coated particles have now been largely confirmed by the AGR program (Section 5.3.2).

5.3.1.3 Fuel Performance Analysis Methods

Computer codes have been developed from past prismatic HTGR programs that model fuel performance and FP release from the fuel. These codes are used for predicting source terms; consequently, these codes are described in [NGNP MST WP 2010] [55]. Current work on the fuel performance model development on the NGNP/AGR Fuel Program is described in Section 5.5.

5.3.2 AGR Program Experience with LEU UCO TRISO in Compacts

The umbrella objective of the AGR program is to qualify LEU UCO TRISO-coated particle fuel for use in modular HTGRs [AGR TDP 2016] [18]. TRISO-coated particles must be fabricated on an industrial scale, as opposed to small batches in a laboratory, for qualification testing. The fuel development and qualification activities under the AGR Fuel Program include fuel manufacturing process and Quality Control methods development, irradiation testing, post-irradiation examination, post-irradiation heating (PIH) tests to bound modular HTGR accident conditions, and fuel performance model development and validation. The program also includes tests to quantify how the fission products are transported through the fuel kernel, particle coatings and the carbonaceous matrix that comprise the fuel compact. Validation of fuel performance and FP transport models – which was part of the original program – is no longer part of the AGR program with the cessation of the NGNP project. All AGR program activities are performed in strict compliance with NQA-1 standards [ASME 2012] [19].

At the start of the AGR Program [e.g., AGR TDP 2008] [84], without a reactor design concept having been selected by the DOE for the NGNP Project from among modular HTGR alternatives, the program decided to qualify fuel to an operating envelope, shown in Figure 25, that would bound both pebble bed and prismatic options. Consequently, the program selected UCO as the fissile kernel of choice because of its ability to limit CO production and kernel migration under irradiation, phenomena that are potentially life limiting for traditional UO_2 TRISO fuels operating at the upper temperature range (~1250°C) and high burnup of a prismatic modular HTGR. The AGR program is organized around a series of irradiation tests that are summarized in Table 9.

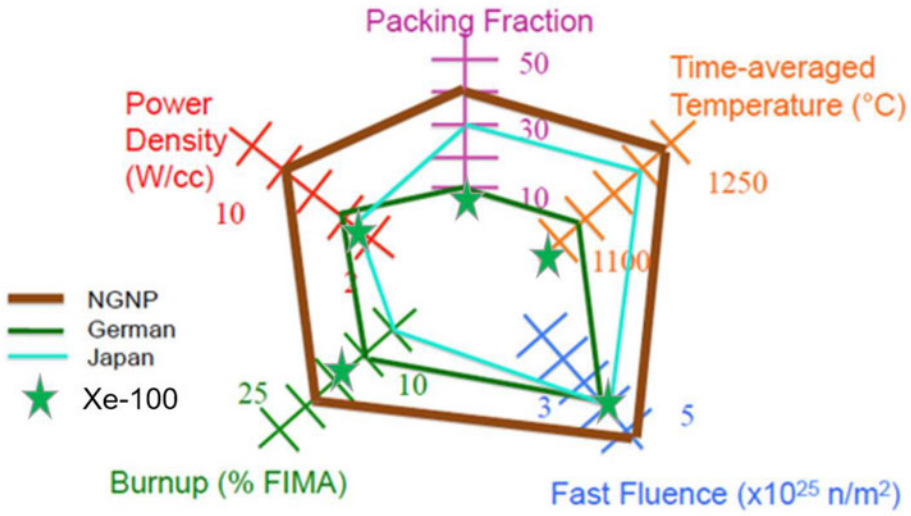

Figure 25: Assumed Performance Envelope for Qualification of AGR Fuel

The preliminary design performance envelope for the Xe-100 core (green star) is comparable to the German envelope shown in the figure and well within the performance envelope for the NGNP that serves as the basis for the AGR Fuel Program.

Table 9. AGR Fuel Program Irradiation Tests.

Capsule	Test Description	Test Objective/Expected Results
AGR-1	**Shakedown Test/Early Fuel Performance Demonstration Test** Contents included compacts made from UCO fuel particles coated in 2-inch laboratory scale coater at ORNL. A baseline fuel particle composite and three variant fuel particle composites were tested. The variants included two particle composites coated using different IPyC coating conditions and one particle composite coated using different SiC coating conditions.	Gain experience with multi-cell capsule design, fabrication, and operation to reduce chances of capsule or cell failures in subsequent capsules. Obtain early data on irradiated fuel performance and support development of a fundamental understanding of the relationship between fuel fabrication process and fuel product properties and irradiation performance. Provide irradiated UCO fuel for accident simulation testing (i.e., heating tests).
AGR-2	**Fuel Performance Demonstration Fuel** Contents included compacts containing UCO particles made in a large coater and UO_2 particles made by B&W, AREVA, and PBMR in different size coaters. AGR-2 had 6 independently monitored and controlled capsules in a test train design essentially the same as demonstrated in AGR-1. One capsule of UCO fuel was operated with a	Provide irradiation performance data for UCO and UO_2 fuel variants and irradiated fuel samples for PIE and post-irradiation heating tests to broaden options and increase prospects for meeting fuel performance requirements and to support development of a fundamental understanding of the relationship between fuel fabrication process and fuel

Capsule	Test Description	Test Objective/Expected Results
	maximum time-averaged temperature of about 1400°C as a performance margin test of the fuel.	product properties and irradiation performance. Also, establish irradiation performance margin for UCO fuel.
AGR-3/4[1]	**Fission Product Transport** Contents included compacts of LEU UCO particles seeded with designed-to-fail (DTF) LEU UCO particles to provide a well-defined FP source. Fuel test element utilized a concentric-ring design to provide a 1-D geometry to facilitate derivation of effective diffusivities. Test capsules operated at different temperatures and operation was maintained isothermal to the extent practical.	Provide data on fission gas release from failed particles, fission metal diffusion in kernels, and gas and metal diffusion in coatings for use in development of FP transport models.
AGR-5/6[2]	**Fuel Qualification** Contents to include a single fuel type made using process conditions and product parameters considered to provide best prospects for successful performance based on process development results and available data from AGR-1 and AGR-2; variations in cell irradiation temperatures per test specification	Provide irradiation performance data for the reference fuel and irradiated fuel samples for PIE and post-irradiation heating tests in sufficient quantity to demonstrate compliance with statistical performance requirements under normal operation and accident conditions.
AGR-7[3]	**Fuel Performance Margin Test** Contents to include same fuel type as used in AGR-5/6 but to be tested under service conditions that exceed the anticipated operating envelop in anticipation that some measurable level of fuel failure would occur (i.e., margin test)	Provide fuel performance data and irradiated fuel samples for PIE and post-irradiation heating tests and PIE in sufficient quantity to demonstrate capability of fuel to withstand conditions beyond AGR-5 and -6 in support of plant design and licensing.
AGR-8[4]	**Fission Product Transport Model Validation** Contents were to have included compacts seeded with LEU UCO with missing buffers or DTF UCO particles. Test geometry had not yet been finalized upon cancellation of the test, but would likely have utilized a multi-fuel stack design to represent prismatic fuel block geometry. Capsules would have operated at different temperatures, and the test operating history would have included temperature cycling.	Provide irradiated fuel performance data and irradiated fuel samples for safety testing and PIE to determine material properties and fission product gas and metal releases from compacts with known quantities of failed particles for use in validation of fuel performance modelling and fission product transport codes.

1. AGR-3 and AGR-4 were originally planned with each test rig containing six independent capsules. The AGR Fuel Program combined them into a single test, currently referred to as AGR-3/4, which is a test rig containing 12 independent capsules.

Capsule	Test Description	Test Objective/Expected Results
2. AGR-5 and AGR-6 were also originally planned as two separate irradiation tests, but it is now combined into a single test. 3. AGR-7 was originally planned to be a fuel performance validation test but was redefined as a margin test after all of the validation work scope was deleted from the program. 4. AGR-8 was cancelled when the NGNP Project was ended.		

A comprehensive status report summarizing the results of the AGR Program to date has been prepared [Petti 2016] [21]; that report is excerpted liberally in the three subsections that follow. Since the AGR program is a critically important element of the planned approach to qualifying the reference Xe-100 fuel, the reader is encouraged to read [Petti 2016] [21] in its entirety. Although the Xe-100 reference fuel uses the specifications of AGR-5/6/7 TRISO particles, the results of the AGR-1/2 irradiation described in the EPRI licensing topical report [1] and associated NRC safety evaluation [7] are also helpful in understanding the performance of all UCO-based TRISO particle fuel.

5.3.2.1 Fabrication of UCO in Compacts

The processes for the fabrication of fuel kernels, TRISO-coated particles, and both fuel compacts and spheres were described in Section 5.1.4. The AGR program has made substantial improvements in the processes for fabricating LEU UCO kernels, TRISO-coated particles, and fuel compacts as summarized below. The optimized processes for kernel fabrication and coating will carry directly over to the production of UCO TRISO-coated particles for use in Xe-100 fuel spheres. The improvements made in as-manufactured fuel characterization methods and in fuel QC techniques will also be applicable to UCO TRISO-coated particles.

[[

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A comprehensive set of fuel product and fuel process specifications were developed by the AGR Fuel Program to assure that the test fuel would have the required as-manufactured fuel attributes necessary to meet in-reactor fuel performance requirements. The fuel specifications implemented by the AGR Program are consistent with the NGNP fuel requirements summarized in Section 4, and they are anticipated to be consistent with the Xe-100 fuel requirements as well. The key AGR fuel product specifications and the attendant QC methods to assure compliance with them are summarized in

.

Table 10. QC Methods for UCO TRISO Fuel in Compacts

Property	QC Method
Kernels	
Uranium content	Wet chemistry
U-235 enrichment	Mass spectrometry
Impurities	Emission spectrometry, wet chemistry, and/or ICP-MS
C/U ratio	Combustion (carbon) and wet chemistry (uranium)
O/U ratio	Combustion (oxygen) and wet chemistry (uranium)
Bulk density	Mercury pycnometry
Diameter	Radiography, ceramography, optical shadow
Coatings	
Missing buffer fraction	Radiography (or screening)
Buffer density	Calculated from particle density and volume (by mercury pycnometry)
Coating thicknesses	Radiography or ceramography
IPyC and OPyC anisotropy	BAF_o by reflection of polarized light
IPyC and OPyC density	Liquid gradient column
SiC density	Liquid gradient column
OPyC crystallite size	X-ray diffraction
OPyC surface connected porosity	Mercury porosimetry
SiC microstructure	Ceramography
Faceting	Radiography, ceramography, optical shadow
Missing or defective OPyC fraction	Optical microscopy
Gold spots (soot inclusions) in SiC coatings	Optical microscopy
Compacts	
Uranium loading	Wet chemistry
Integrity and dimensions	Visual inspection and manual gauging
Matrix density	Calculation
Heavy metal contamination	Acid leach or gaseous HCl leach
Defective SiC coating fraction	Burn-leach

Defective IPyC coating fraction (fuel dispersion)	Compact deconsolidation/Radiography
Defective OPyC coating fraction	Compact deconsolidation/optical microscopy
Impurities	ICP-MS

Table 11, Table 12 and Table 13 provide the typical attributes of UCO kernels, TRISO-coated particles, and fuel compacts, respectively, used for the AGR-2 irradiation. The product specifications were met at 95% confidence with the exception of the HM contamination in the finished compacts which slightly exceeded the specification (2.4 x 10-5 > 2.0 x10-5).

Table 11. UCO Kernel Attributes for AGR-2 Fuel

UCO Kernel Property	Specified Range for Mean Value	Actual Mean Value ± Population Standard Deviation
Diameter (µm)	425 ± 20	426.7 ± 8.8
Density (Mg/m^3)	≥10.3	10.966 ± 0.033
U-235 enrichment (wt%)	14.0 ± 0.10	14.029 ± 0.026
Carbon/uranium (atomic ratio)	0.40 ± 0.10	0.392 ± 0.002
Oxygen/uranium (atomic ratio)	1.50 ± 0.20	1.428 ± 0.005
[Carbon + oxygen]/uranium (atomic ratio)	≤2.0	1.818 ± 0.005
Total uranium (wt %)	≥88.5	89.463 ± 0.051
Sulfur impurity (ppm – wt)	≤1500	365 ± 12
Phosphorus impurity (ppm – wt)	≤1500	≤30
All other impurities (ppm-wt)	≤100	Below minimum detection limits and within specification

Table 12. Properties of TRISO-coated UCO Particles for AGR-2 Fuel

Property	Specified Range for Mean Value	Actual Mean Value ± Population Standard Deviation
Buffer thickness (µm)	100 ± 15	98.9 ± 8.4
IPyC thickness (µm)	40 ± 4	40.4 ± 2.5
SiC thickness (µm)	35 ± 3	35.2 ± 1.2
OPyC thickness (µm)	40 ± 4	43.4 ± 2.9 [a]
Buffer density (Mg/m^3)	1.05 ± 0.10	Not measured [b, d]
IPyC density (Mg/m^3)	1.90 ± 0.05	1.890 ± 0.011

SiC density (Mg/m^3)	≥3.19	3.197 ± 0.004
OPyC density (Mg/m^3)	1.90 ± 0.05	1.907 ± 0.007
IPyC anisotropy (BAF)	≤1.045	1.0349 ± 0.0012
OPyC anisotropy (BAF)	≤1.035	1.0263 ± 0.0011
IPyC anisotropy post compact anneal (BAF)	Not specified	1.0465 ± 0.0049
OPyC anisotropy post compact anneal (BAF)	Not specified	1.0429 ± 0.0019
SiC sphericity (aspect ratio)	Mean not specified [f]	1.037 ± 0.011
OPyC sphericity (aspect ratio)	Not specified	1.052
Particle diameter [g] (µm)	Mean not specified	873.2 ± 23
Particle mass (mg)	Mean not specified	1.032 ± 0.003

a. 95% upper confidence thickness exceeds specifications. Justification of acceptance: OPyC thickness does not affect the compacting process or the fuel performance during irradiation (BWXT 09/2008 and 2009).
b. BWXT's hot sampling system did not allow both buffer and IPyC density measurements (BWXT 2009).
c. Single determination, no statistical confidence available (BWXT 2009).
d. Similar samples showed measurement results within specifications (BWXT 09/2008 and 2009).
e. Lower confidence level.
f. Critical region is specified such that ≤1% of the particles shall have an aspect ratio ≥1.14 for UCO fuel.
g. Based on mean average particle measurements, not sums of mean layer thicknesses.

Table 13. Selected Properties for AGR-2 Compacts.

Property	Specified Range for Mean Value	Actual Mean Value ± Population Standard Deviation
Compact mass (g)	Not specified	6.294 ± 0.011
Mean uranium loading (g U/compact)	1.265 ± 0.07 (UCO) 1.00 ± 0.05 (UO_2)	1.257 ± 0.03
Diameter [(b)] (mm)	12.22 – 12.46	12.286 ± 0.005
Length [(b)] (mm)	25.02 – 25.40	25.141 ± 0.017
Number of particles per compact [(a)]	Not specified	3176
Particle volume packing fraction (%)	Not specified	37
Effective overall compact density[(a)] (Mg/m^3)	Not specified	2.11
Compact matrix density (Mg/m^3)	≥1.45	1.589 ± 0.005
Compact weight % U [(a)]	Not specified	19.97
Compact weight % O [(a)]	Not specified	1.92
Compact weight % Si [(a)]	Not specified	6.85
Compact weight % C [(a)]	Not specified	71.26

Iron content (µg Fe outside of SiC/compact)	≤25	4.04
Chromium content (µg Cr outside of SiC/compact)	≤50	0.61
Manganese content (µg Mn outside of SiC/compact)	≤50	0.136
Cobalt content (µg Co outside of SiC/compact)	≤50	1.115
Nickel content (µg Ni outside of SiC/compact)	≤50	0.96
Calcium content (µg Ca outside of SiC/compact)	≤50	39.34
Aluminum content (µg Al outside of SiC/compact)	≤50	29.60
Titanium content (µg Ti outside of SiC/compact)	Note (c)	2.81
Vanadium content (µg V outside of SiC/compact)	Note (c)	17.09
U contamination fraction [(d)] (g exposed U / g U in compact)	$\leq 2.0\times10^{-5}$	$\leq 2.4\times10^{-5}$ [(e)]
U contamination fraction w/o exposed kernels (g leached U / g U in compact)	Not specified	1.59×10^{-6}
Defective SiC coating fraction [(d)]	$\leq 1.0\times10^{-4}$	$\leq 1.2\times10^{-5}$
Defective IPyC coating fraction [(d)]	$\leq 1.0\times10^{-4}$	$\leq 4.7\times10^{-5}$
Defective OPyC coating fraction [(d)]	$\leq 1.0\times10^{-2}$	$\leq 9.4\times10^{-4}$

a. Approximate calculated value derived from other characterized properties.
b. Allowable range corresponding to upper and lower critical limits specified with no compacts exceeding the limits, which require 100% inspection of all compacts.
c. Mean value specification of ≤ 400 µg Ti plus V outside SiC/compact.
d. Value is an estimate of an attribute property, not the mean of a variable property.
e. Values represent 95% confidence levels and exceed specification.

5.3.2.2 Irradiation Results

The objective of the AGR fuel irradiation program is to provide data on UCO TRISO fuel performance under normal operating conditions, initially to demonstrate and then to qualify fuel for operation in a modular HTGR. The objectives of these tests are to provide irradiation performance data to support fuel process development, to qualify fuel for normal operating conditions and accident conditions, and to support development and validation of fuel performance and FP transport models. A further underlying objective of the AGR Fuel Program is to further the development of a fundamental understanding of the relationship between the fuel fabrication process, fuel product properties, and irradiation and accident condition

performance, which will lead to improved models for predicting the performance of the fuel in a modular HTGR. The AGR irradiation test program is summarized in Table 9 [AGR TDP 2016] [18].

5.3.2.2.1 AGR-1 and AGR-2 Tests

The experiment test train for AGR-1 and AGR-2 tests consisted of six separate stacked capsules vertically centered in the ATR core. Each capsule had its own custom blended gas supply and exhaust for independent temperature control and R/B monitoring. Temperature control of the capsules was accomplished by adjusting the mixture ratio of two gases with differing thermal conductivities to control the heat transfer across an insulating gas jacket between the heat source (fuel fissions and gamma heating of capsule materials) and the relatively cold reactor coolant (light water at 52°C). Helium was used as the high (thermally) conductive gas, and neon was used as the insulating gas.

A horizontal capsule cross-section at the top of the test train is shown in Figure 26. The capsules were approximately 35 mm (1-3/8 inches) in diameter and 150 mm (6 inches) in height, including the plenums between adjacent capsules. Each capsule contained 12 prototypical right circular cylinder fuel compacts nominally 12.3 mm (1/2 inch) in diameter and 25 mm (1.0 inch) long. For AGR-1, the fuel contained TRISO-coated 350 µm UCO particles with an enrichment of 19% U-235[13] [Collin 2015a] [100]. Fuel compacts are identified by their location in the test train using a three-digit (X-Y-Z) nomenclature, where X refers to the capsule number, Y refers to the axial level within the capsule (Level 4 is at the top of the capsule and Level 1 is at the bottom), and Z refers to the stack number [EPRI-AR-1] [1].

For AGR-2, two different types of fuels were tested [Collin 2014] [102]. The first fuel was TRISO-coated 425 µm UCO particles with an enrichment of 14% U-235. The second fuel was TRISO-coated 500 µm UO_2 particles with an enrichment of <10% U-235. The LEU UO_2 fuel was manufactured by three different organizations: (1) BWXT, (2) CEA and (3) PBMR. Each fuel manufacturer had a dedicated capsule within the six-capsule test train as indicated in Table 14.

[13]The particles fabricated for AGR-1 were based on the fissile particle in a two-particle fissile/fertile design by GA. Subsequently, the NGNP Project moved to a single particle design and developed the 425 µm/14 % enriched particle. For AGR-5/6/7 the same particle size will be used but with the enrichment increased slightly to 15.5% based on design studies conducted at GA prior to the end of the NGNP Project [CDR 2010] [58].

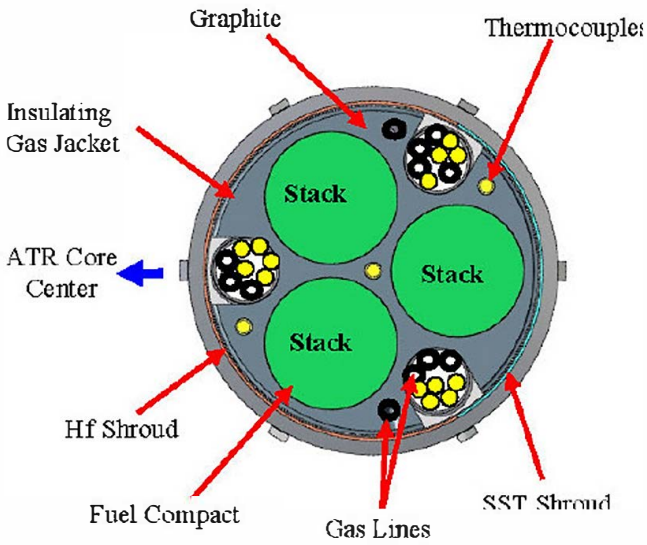

Figure 26: Horizontal Cross-section of an AGR-1/-2 Experiment Capsule

Table 14. AGR-2 Test Fuel and Irradiation Conditions

Capsule	Source	Fuel Type	Burnup (% FIMA)	Fast Fluence (10^{-25} n/m^2)	Time-Ave. Temperature (°C)	
					Vol. Ave.	Peak
6	BWXT	425 µm UCO	10.8	2.7	1074	1183
5	BWXT	425 µm UCO	12.9	3.4	1101	1210
4	PBMR	500 µm UO_2	PBMR Proprietary Information			
3	BWXT	500 µm UO_2	10.7	3.5	1032	1105
2	BWXT	425 µm UCO	13.2	3.5	1252	1360
1	CEA	500 µm UO_2	CEA Proprietary Information			

To monitor fission gases, the system routed the outlet gas from each capsule to individual FP monitors, as shown in Figure 27. The FP monitors consisted of a spectrometer for identifying specific fission gases and a gross gamma detector to provide indication when a pulse release of fission gases passed through the monitor.

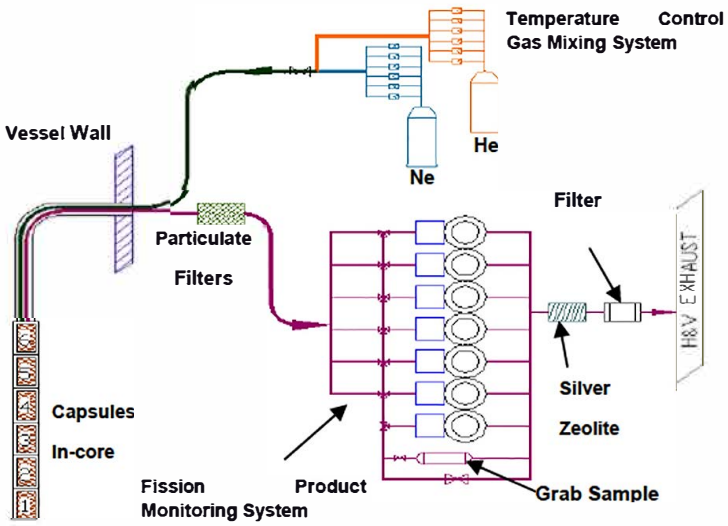

Figure 27: AGR-1/-2 Purge Gas Flow Path

Fission gas release measurements from the AGR-1 and AGR-2 experiments are compared to historic German and U.S. irradiations in Figure 28.[14] The gas release for AGR-1 was extremely low indicating no particle failures during the irradiation [Collin 2015a] [100]. Thus, AGR-1 is the best irradiation performance of a large quantity of TRISO fuel ever achieved in the U.S., even though the UCO fuel substantially exceeded the German burnup levels. These results have confirmed the expected superior irradiation performance of UCO at high burnup in that no kernel migration, no evidence of CO attack of SiC, and no indication of severe SiC attack by noble metal or lanthanide fission products has been observed. Zero fuel failures in AGR-1 translates into a 95% confidence failure fraction of $<1 \times 10^{-5}$, a factor of 20 better than the prismatic reactor design in-service failure fraction requirement of 2×10^{-4} (95% confidence limit).

[14]The fission gas release rates are expressed traditionally as a release (R/B). At steady-state, the R/B and fractional release are equal.

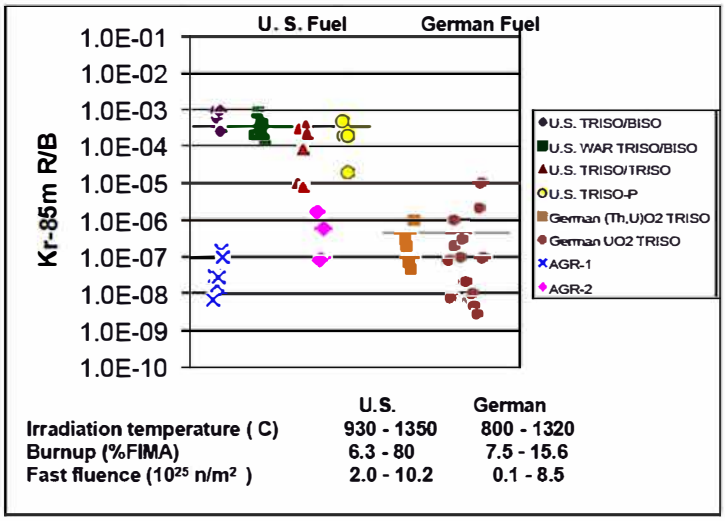

Figure 28: Kr-85m R/B Data for AGR-1/-2 Compared to Historical U.S. and FRG Data

The R/B data for AGR-2 shown in Figure 28 are higher than AGR-1 and indicate one exposed kernel in each UCO capsule based on the level of heavy metal contamination measured during QC of the fuel [Collin 2014] [102]. Detailed analysis of the three UCO test capsules [Einerson 2016] [103], and PIE results suggest that either zero particles or one particle may have failed in one UCO capsule based on fission gas release. Because of gas flow problems in the irradiation capsule that occurred about one-third of the way into the experiment, it is difficult to determine precisely. Thus, assuming zero or one particle failure out of 114,000 UCO TRISO-coated particles, a 95% confidence failure fraction between 2.4×10^{-5} and 4.2×10^{-5} is obtained, which is a factor of about five to eight below the designer specification of 2×10^{-4} (95% confidence limit) [Hanson 2009] [104].

The AGR-1 irradiation resulted in UCO TRISO fuel being exposed to very high temperatures for long times, well in excess of those expected in an actual modular HTGR. By comparison, peak time average temperatures in prismatic modular HTGRs are usually less than 1250°C [e.g., Hanson 2009] [104]. The more severe AGR-1 irradiation conditions, compared to the vast majority of historic modular HTGR designs, suggest substantial fuel performance margin. This performance margin was further demonstrated by the results for Capsule 2 in the AGR-2 test which was designed to operate at a time averaged peak temperature of 1400°C (i.e., to be an early margin test).

5.3.2.2.2 AGR-3/4 Test

The objective of the AGR-3/4 test is very different from that of AGR-1 or AGR-2 [Collin 2015c] [105]. It is devoted to understanding FP transport behavior in TRISO-coated particles and the carbonaceous components (fuel-compact matrix and fuel-element graphite) in a prismatic HTGR core. Thus, it supports the HTGR source term evaluation and not TRISO-coated particle fuel performance assessment. The fuel irradiated in each AGR-3/4 capsule contained conventional TRISO driver fuel particles and designed-to-fail (DTF) fuel particles. The UCO kernels of conventional fuel particles were similar to the baseline fuel used in the AGR-1 experiment. The DTF fuel particles contained kernels identical to the driver fuel kernels, but their coatings were designed to fail early under irradiation to provide a well-defined fission product source.

The AGR-3/4 test, the most complex irradiation experiment performed in the AGR series, was comprised of 12 independently controlled and monitored capsules stacked on top of each other to form the test train using the full 1.22-m active core height as shown in Figure 29. Each capsule contained four 3.81-cm long compacts. Each fuel compact contained about 1,872 conventional UCO driver fuel coated particles and 20 DTF UCO fuel particles.

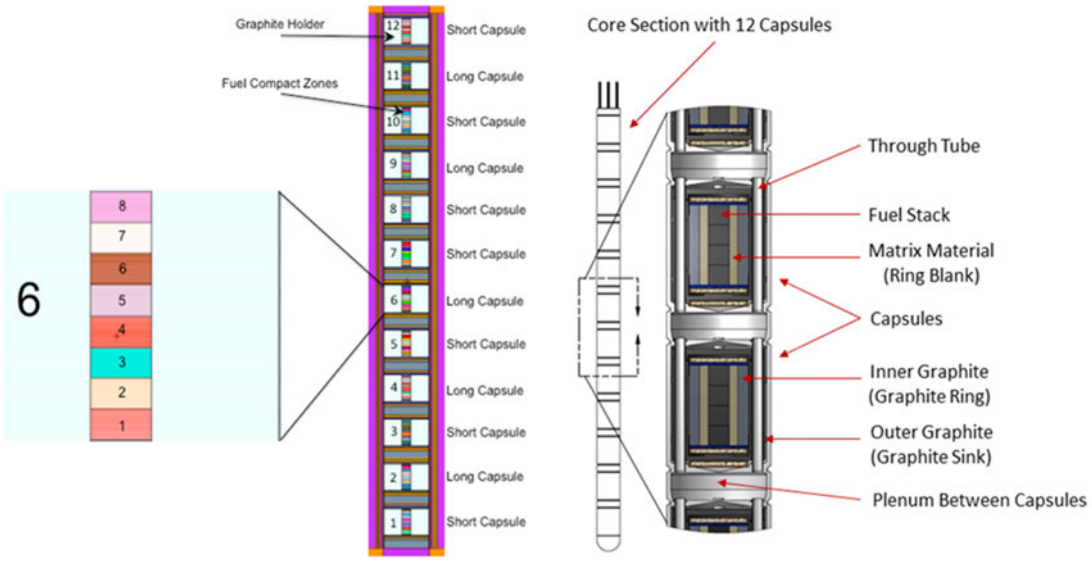

Figure 29: AGR-3/4 Cross-Section (left) and an AGR-3/4 Capsule (right)

A key objective of the AGR-3/4 test was to measure the release of fission gases from failed UCO particles in terms of the R/B per failed particle as a function of temperature, burnup and half-life to establish a correlation that can be used by HTGR designers.[15] Because AGR-2 also had one exposed particle in each capsule, that release data can be compared to the measurements from AGR-3/4. In addition, there are a handful of historic irradiations that had very limited but useful data for this comparison. This large amount of R/B data from AGR-2 and AGR-3/4 irradiations allows assessment of the effect of isotopic decay constants and fuel temperatures on FP release. Figure 30 plots the R/B per failed particle data from the AGR-3/4 capsules and AGR-2 capsules and a best fit of the aggregate data [Binh 2016] [101]. The R/B per failed particle for both krypton and xenon isotopes are less than 1% and are not temperature sensitive for fuel temperatures <~1050°C. However, when the fuel temperature is greater than ~1050°C, the R/B per failed particle increases exponentially with increasing fuel temperature.

[15]There are other FP transport objectives in AGR-3/4 including effective diffusion coefficients for metallic fission products in UCO kernels and transport coefficients in compact fuel matrix and fuel element graphite. However, these data are not yet available from PIE, which has been underway for about several years as of this writing.

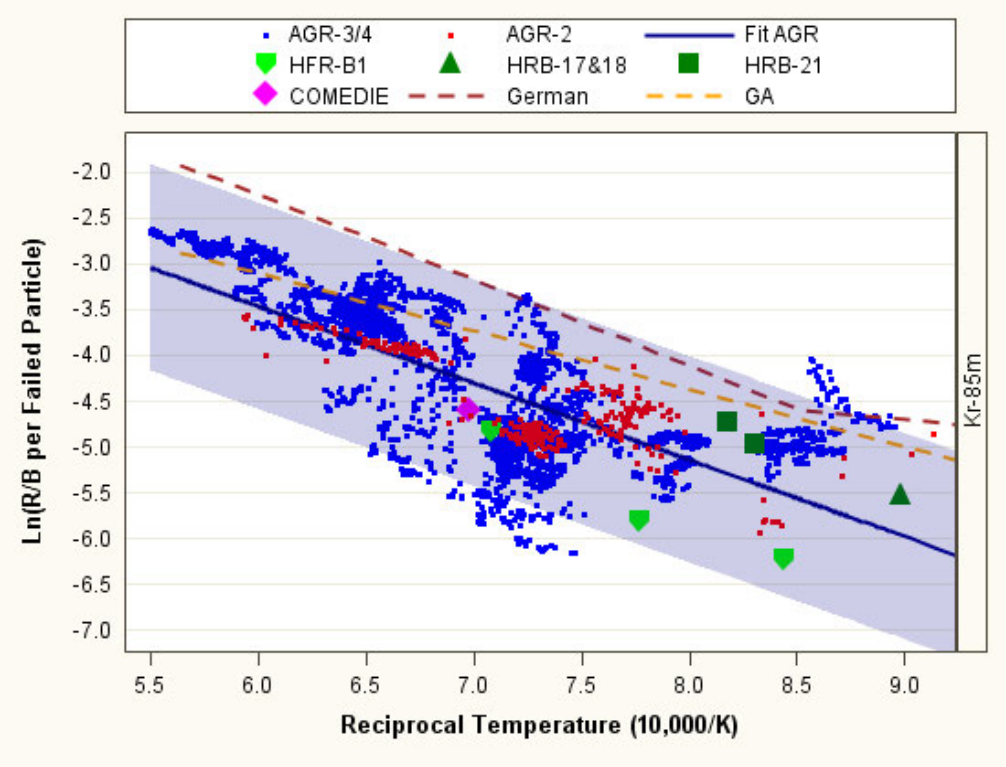

Figure 30: R/B per Failed Particle Data from AGR 2 and 3/4 Tests and Historical Data

5.3.2.2.3 AGR-5/6/7 Test

The AGR-5/6/7 test [Petti 2016] [21] is both a fuel qualification irradiation (AGR-5/6) and a margin irradiation (AGR-7) for the industrially-produced UCO TRISO-coated particle fuel developed by the AGR program [Marshall 2016] [106] and [Maki 2015] [22]]. Unlike the previous AGR irradiations, the test train for the AGR-5/6/7 experiment contains five capsules (Figure 31) with different combinations of irradiation temperature and fuel burnup that more broadly span the temperature and burnup range expected in a modular HTGR to provide more representative data on TRISO fuel performance. For AGR-5/6, 30% of the particles were to operate <900°C, 30% to operate at 900°C-1050°C, 30% to operate at 1050°C to 1250°C, and the remaining 10% to operate at 1250°C to 1350°C. For the margin test, AGR-7, all 50,000 particles were to operate at 1350°C to 1500°C. AGR-5/6/7 utilized the full 1.2-meter active core height in ATR to provide the desired broad range of fuel burnup and temperature combinations. As shown in Table 15 [21], the fuel compact burnup goals are a minimum of 6% FIMA and a maximum of 18% FIMA with a planned duration of 500 effective full power days (EFPDs).

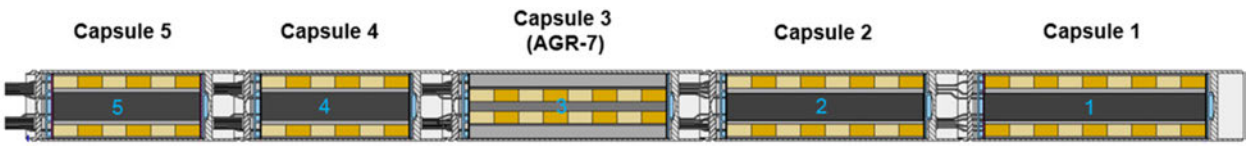

Figure 31: Schematic view of the AGR-5/6/7 test train (Note: Capsule 5 is at the top of the test train).

The irradiation commenced on February 16, 2018 in the north-east flux trap of INL's ATR. Unexpected measurements of fission product gases, characteristic of in-pile particle failure, were detected in at least two of test capsules and the program decided to end the irradiation test on July 22, 2020 in order to commence forensic examination of the experiment and determine the cause of the unexpected behavior. AGR-5/6/7 compacts were actually irradiated for approximately 360.9 EFPDs over nine ATR cycles. R/B values were stable in the 10^{-8} – 10^{-6} range during the first five cycles of the experiment, and no in-pile particle failures were observed based on gross gamma counts through that time [Binh 2020] [107].

Table 15. AGR-5/6/7 Irradiation Test Specifications

Parameter	AGR-5/6 Specification	AGR-7 Specification
Instantaneous peak temperature for each capsule (°C)	≤1800	≤1800
Time average temperature distribution goals (°C)	≥600 and <900 for about 30% of fuel ≥900 and <1050 for about 30% of fuel ≥1050 and <1250 for about 30% of fuel ≥1250 and <1350 for about 10% of fuel	Not specified
Time average, peak temperature goal (°C) (for one element)	1350 ± 50	1500 ± 50
Time average, minimum temperature goal (°C)	≤700	Not specified
Minimum compact average burnup (% FIMA)	>6 for all compacts	>6 for all compacts
Maximum fuel compact average burnup (% FIMA)	>18 for at least one compact	>18 for at least one compact
Maximum fuel compact fast neutron fluence (n/m², E > 0.18 MeV)	≥ 5.0×10^{25} for at least one compact and ≤ 7.5×10^{25} for all compacts	≥ 5.0×10^{25} for at least one compact and ≤ 7.5×10^{25} for all compacts
Minimum fuel compact fast neutron fluence (n/m², E > 0.18 MeV)	> 1.5×10^{25}	> 1.5×10^{25}
Instantaneous peak power per particle (mW/particle)	≤400	≤400

Figure 32 plots the R/B per failed particle data from the AGR-5/6/7 capsules and a best fit of the aggregate data [Binh 2019] [108].

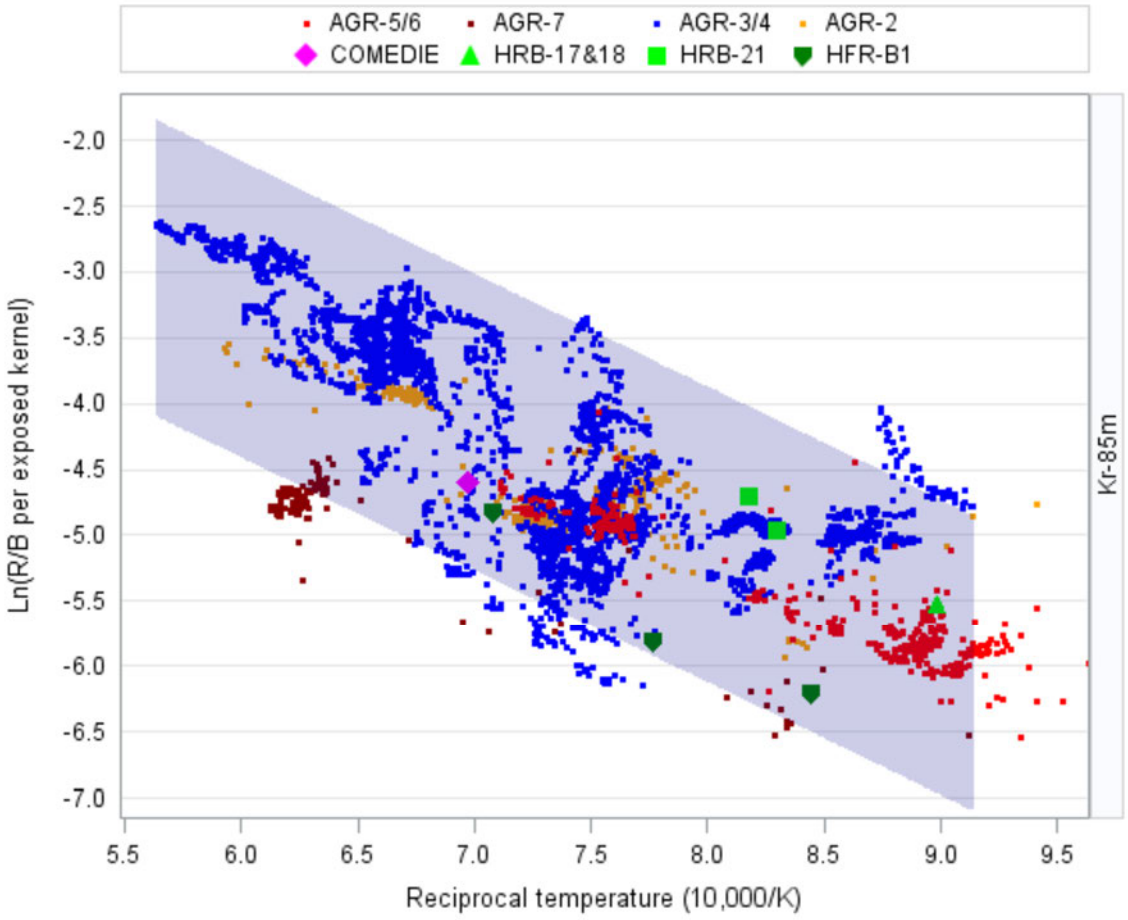

Figure 32: R/B per Failed Particle Data from AGR 2, 3/4, & 5/6/7 Tests and Historical Data

5.3.2.3 PIE and Safety Testing

The objective of the PIE and safety testing is to characterize the performance of TRISO fuel after irradiation and during postulated accident conditions. These activities also support the fuel development effort by providing feedback on the performance of kernels, coatings, and compacts. Data from PIE and accident testing in combination with the in-reactor measurements provide the data necessary to demonstrate compliance with fuel performance requirements and to support the development and validation of computer codes. At this writing, the PIE of UCO TRISO fuel irradiated in AGR-1 is complete, and similar work for AGR-2 is underway. The PIE was focused on evaluating fuel performance during irradiation and during post-irradiation high temperature heating tests in helium. Key aspects of fuel performance that were investigated were FP release from particles and compacts, and radiation-induced changes in kernel and coating microstructures. Safety tests were performed by heating the fuel compacts in helium at temperatures of 1600, 1700, or 1800°C, with nominal hold times of 300 hours.

5.3.2.3.1 Fission Product Distributions

A mass balance of fission products was performed by quantifying the FP content in the various components of all six irradiation capsules in AGR-1 (representing the inventory released from the fuel compacts during irradiation) and within the matrix of eight compacts (representing the inventory released from particles but retained in the compact outside of the SiC layer during irradiation). The results are presented graphically in Figure 33, showing the range of inventory fractions (based on either single compacts or all twelve compacts in a single capsule, as appropriate) found in the compacts and the irradiation capsules. Very low release of key metallic fission products (except silver) confirms the excellent performance measured under irradiation and demonstrates the robustness of the SiC layer as a barrier to FP release.

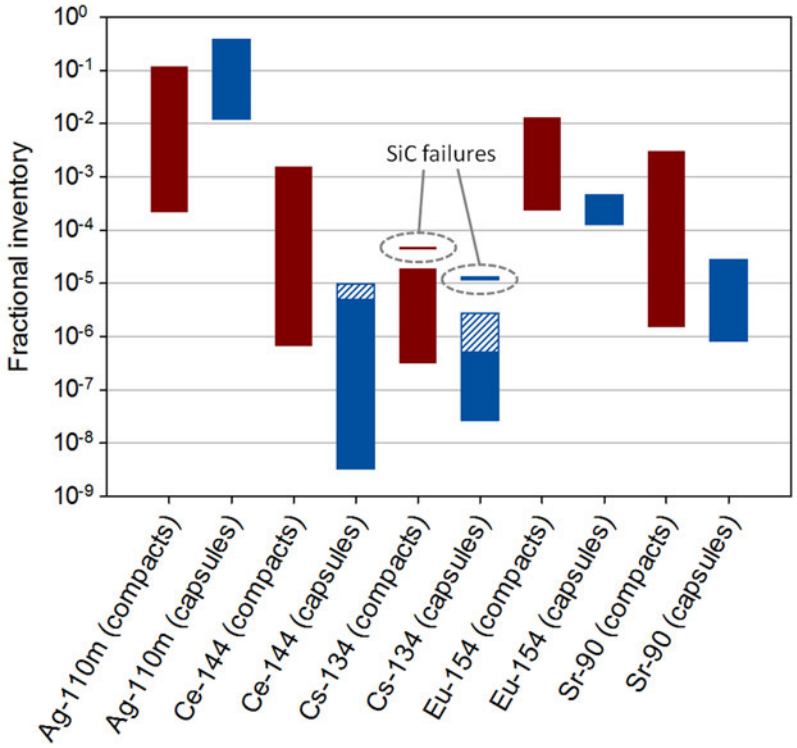

Figure 33: Fission Product Mass Balance for AGR-1 Capsules

As indicated in Figure 33, the cesium release from the fuel was very low. In compacts with no particles exhibiting a failed SiC layer, the Cs-134 fractional inventory in the matrix was $<2 \times 10^{-5}$, and in capsules containing zero particles with failed SiC, the Cs-134 fractional release from the compacts was $<3 \times 10^{-6}$. Releases of Sr and Eu into the compact matrix were generally small ($\sim 1 \times 10^{-6}$ to 1×10^{-2}), indicating some diffusion through the coatings. Ce-144 releases to the compact matrix were a factor of 10 smaller ($\sim 1 \times 10^{-6}$ to 1×10^{-3}). However, the amount of these materials in the capsule components (graphite holder, end caps, and steel shell) tends to be much less compared to what is present in the compact matrix, demonstrating good retention in the matrix.

The level of Pd found outside the SiC was approximately 1% in the five compacts analyzed. Despite this large amount of Pd in the fuel matrix, no widespread Pd corrosion or attack of SiC has been observed during metallographic examination of a number of as-irradiated TRISO-coated particles. This was unexpected since Pd attack of SiC at high burnup in TRISO fuel has been postulated as a potential fuel particle failure mode [Maki 2007] [112].

Silver release was high and varied significantly from compact to compact as a result of the differences in the temperatures experienced by each compact under irradiation.

5.3.2.3.2 Irradiated Fuel Particle Microstructural Evaluation

Extensive microscopic examination of particle cross-sections was performed. This included numerous cross-sections of select as-irradiated compacts [Ploger 2014] [113] as well as loose particles deconsolidated from numerous as-irradiated or safety-tested compacts [Demkowicz 2015] [114]. In addition, a select number of particles were analyzed with nondestructive x-radiography with 3D tomographic reconstruction. Common features in the irradiated particles included densification of the buffer layer and swelling of the kernel with related formation of gas-filled bubbles (Figure 34). There was no detectable high-burnup kernel migration, indicating the efficacy of the UCO fuel in limiting the oxygen partial pressure in the fuel and the formation of carbon monoxide.

In the majority of the AGR-1 particles, the buffer layer debonded from the IPyC layer. This was observed as either complete (Figure 34a) or partial (Figure 34b) debonding in the polished plane analyzed. Much less common were particles in which the buffer and IPyC layers remained completely bonded in the plane observed (Figure 34c), where the buffer densification resulted in the inner diameter increasing while the kernel swelled to filled the increasing volume. Such particles constituted 4% of approximately 1,000 particles observed in a study of compact cross sections.

Fracture of the buffer layer was not uncommon (observed in 23% of particles studied in the compact cross sections), and these were often accompanied by expansion of the kernel into the gap formed at the point of fracture (Figure 34d). While particles with the representative microstructure shown in Figure 34d were fairly common, there appeared to be no obvious detrimental effects on the outer, dense coating layers, even in cases where the kernel was in direct contact with the IPyC layer. On the other hand, if the buffer-IPyC interface remained intact as in Figure 34c, fracture of the buffer layer was always accompanied by fracture of the IPyC layer, and often also included debonding of the IPyC from the SiC layer (Figure 34e).

Fracture of the buffer layer was not always necessary for IPyC fracture to occur. In some particles, partial debonding of the buffer-IPyC layer apparently led to development of sufficient stress in the IPyC layer to cause fracture (Figure 34f), often with resultant debonding between the IPyC and SiC layers and in rare cases, partial fracture of the SiC at the IPyC-SiC interface (as shown in Figure 34f) that did not lead directly to SiC failure. Because partial buffer-IPyC debonding (Figure 34b) was much more common than no debonding (Figure 34c), this type of IPyC fracture was more common than the type shown in Figure 34e. Subsequent detailed analysis of particles with failed SiC layers [Hunn 2016] [110] revealed that most exhibited buffer-induced IPyC fracture like that shown in Figure 34e or Figure 34f. This fracture then allowed concentration of fission products at the inner SiC surface in the vicinity of the IPyC-SiC debonding site, which subsequently attacked the SiC layer, eventually penetrating the layer and causing a loss in

retention of fission products such as cesium. An important conclusion from this analysis is that low-stress buffer-IPyC debonding appears to be a desirable condition.

Similar behavior (i.e., fracture of the IPyC layer driven by buffer shrinkage coincident with strong buffer-IPyC bonding) was observed during the PIE of AGR-2 particles, but the frequency of occurrence was lower. It is believed that the lower frequency is due in part to a reduced tendency for buffer-IPyC bonding in the AGR-2 particles as a result of minor process variations during buffer and IPyC deposition. In addition, a notable decrease in buffer fracture was observed in AGR-2 Capsule 2, believed to be due to the higher irradiation temperatures driving stress reduction in the layer via thermal creep.

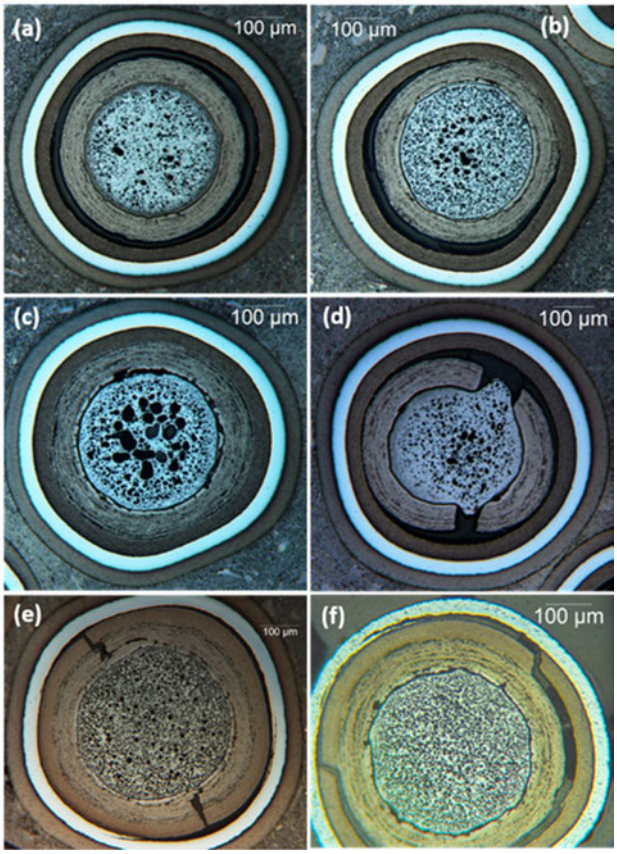

Figure 34: Examples of Various AGR-1 Irradiated Particle Microstructures

No significant dependence of particle morphology on burnup, fast fluence, or temperature was apparent over the range of irradiation conditions represented by the compacts examined in the AGR-1 PIE (burnup of approximately 11 to 19% FIMA). After safety testing, similar particle morphologies were generally observed, although instances of SiC failure increased.

5.3.2.3.3 Safety Testing

Post-irradiation accident simulation heatup testing (“safety testing”) of UCO TRISO-coated particle fuel from AGR-1 has demonstrated excellent robustness of the UCO TRISO fuel. An example of FP release from 1600°C safety testing is shown in Figure 35 [Baldwin 2014] [115]. Very low releases have been measured

in safety testing after hundreds of hours at 1600 and 1700°C [Morris 2016] [62]. Summary results for Cs-134 for 15 AGR-1 compacts are shown in Figure 36, which includes results from tests conducted at 1600, 1700 and 1800°C. During safety testing, compact Cs-134 release fractions were $<5 \times 10^{-6}$ if all SiC layers remained intact. This was true in one case even at 1800°C for ~100 hours, after which a SiC layer failure occurred with corresponding cesium release.

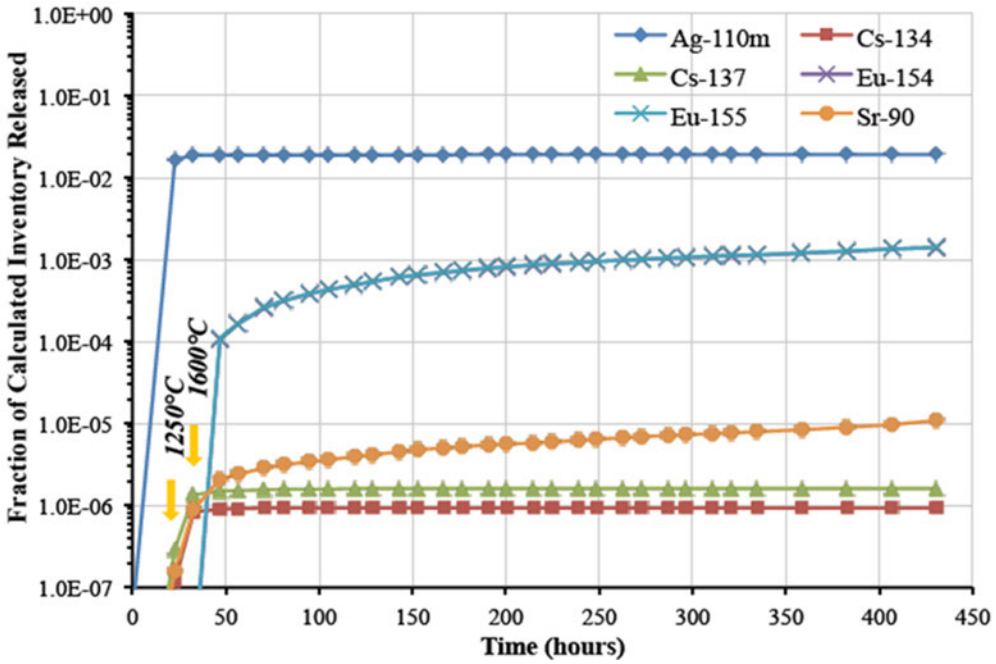

Figure 35: Fission Product Release from AGR-1 Compact 6-4-3 at 1600°C.

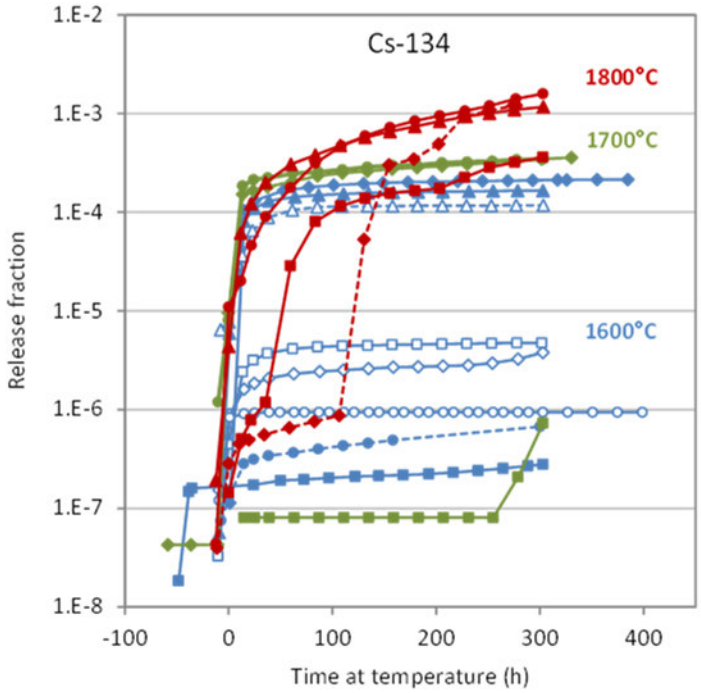

Figure 36: Cs-134 Release from AGR-1 Compacts at 1600, 1700 and 1800°C

No full TRISO-coated particle failures (i.e., no measured Kr-85 release) were observed in accident heating tests of AGR-1 compacts at 1600 or 1700°C. This corresponds to a 95% confident failure fraction of ~7 x 10^{-5}, a factor of eight below the HTGR prismatic reactor specification. Fission product release from UCO TRISO fuel is very low at 1600, 1700, and 1800°C unless a SiC layer fails due to either a manufacturing defect or a radiation-induced failure mechanism. Cs releases after 300 hours at 1600°C are <1 x 10^{-5} from compacts containing only intact particles. For cases where a SiC failure occurs, releases are around 1 to 2 x 10^{-4} (100% release from one particle corresponds to a release fraction of 2.4 x 10^{-4}) and are dependent on the total number of SiC failures that occurred. The best estimate of SiC failures observed during heating at 1600°C is 3 out of about 33,000 particles, which corresponds to ≤2.4 x 10^{-4} at 95% confidence.

A seminal result of the AGR fuel program has been an unequivocal demonstration of the superior high burnup and high temperature performance capability of UCO TRISO fuel compared to LEU UO_2 TRISO fuel as illustrated in Figure 37. The significantly higher Cs-134 release (red curves in the figure) with the UO_2 fuel is an indication of SiC coating degradation, most likely as a result of CO corrosion.

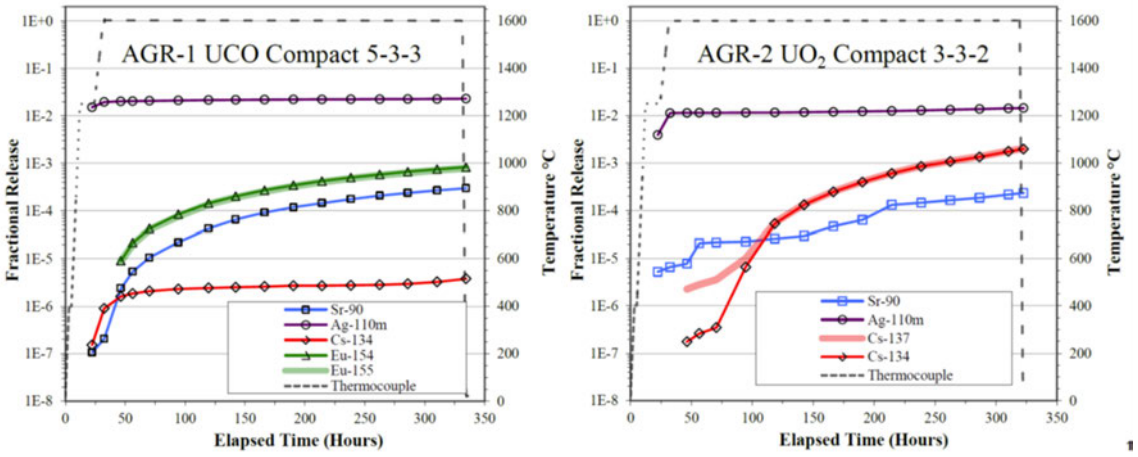

Figure 37: Comparison of UCO and UO_2 Performance in AGR Safety Tests

5.3.2.4 Licensing Implications

The results of the AGR Program have been very successful to date, and they support current safety design and analysis assumptions about fuel performance and radionuclide retention required under the U.S. modular HTGR licensing strategy.

Fabrication of high-quality low-defect fuel is achievable at industrial scale. There is an improved understanding of the TRISO fuel fabrication process and improved fabrication and characterization of TRISO fuel produced by the fuel vendor. Defect fractions on the order of 10^{-5} are achievable. The process produces narrow distributions of fuel attributes (e.g., standard deviation of coating thicknesses and densities are small). The process is stable and repeatable at industrial scale (batch to batch variation is very small).

The AGR program has demonstrated excellent irradiation performance of a large statistical population of UCO TRISO fuel particles under high burnup, high temperature modular HTGR conditions (both AGR-1 and AGR-2). This outstanding performance would result in low FP release into the reactor helium.

Safety testing at 1600°C and 1700°C demonstrated robustness of UCO TRISO under depressurized conduction cooldown conditions. No full TRISO-coated particle failures have been observed at 1600°C or 1700°C. Release of cesium from intact particles at 1600°C was <5 x 10^{-6}. When a SiC layer in a particle failed, some of the Cs from that particle was released. Releases of Ag, Sr, and Eu at 1600°C and 1700°C are attributed to diffusion of these fission products into the fuel matrix during irradiation and subsequent release from the matrix upon high temperature heating. Overall, the results indicate low incremental release of safety-relevant fission products under accident conditions. The results obtained to date from AGR-2 UCO fuel produced at engineering scale are similar to that from AGR-1 lab scale fuel.

Collectively, these irradiation and safety testing results provide a high level of confidence that the AGR fuel program will successfully demonstrate the superior performance capability of UCO TRISO fuel required by the modular HTGR safety design basis.

5.4 EFFECTS OF OXIDIZING ATMOSPHERES ON TRISO-COATED PARTICLE FUEL

Almost all of the post-irradiation heating tests of TRISO fuel that have been done over the past four decades have been performed in pure dry helium. Under these conditions, TRISO fuel, especially the AGR UCO TRISO fuel, has performed well. The only exception to the inert He tests are the bounding tests done by Dr. Schenk in the KORA furnace at KFA Juelich with German LEU UO_2 in loose particles and in AVR spheres heated in water and in air. As summarized in Table 16, reproduced from [TECDOC-978 1997] [69], TRISO-coated particles do not perform nearly as well in oxidizing atmospheres, especially in air, as they do in inert He.

Table 16: KORA PIH Tests of LEU UO_2 in Air

Fuel sample	No. of par-ticles	Burnup [%FIMA]	Test conditions			Kr-85 release		
			Heatup [h]	Max. temp. [°C]	Time [h]	1st failed particle after	No. of failed par-ticles	Fraction of failed particles
92/29. 12	10	9.2	14	1400	400	397 h	1	0.1
73/8. 11	10	4.7	15	1500	25	8 h	10	1
92/29. 13	10	9.2	15	1500	25	3 h	10	1
92/29. 11	10	9.2	28	1620	1	at 1613 °C	10	1
AVR 89/12	16,400	9.4	13	1300	410	258 h	4	$2.4*10^{-4}$
AVR 92/22	16,400	8.8	14	1400	140	1 h	20	$1.2*10^{-3}$
AVR 89/14	16,400	9.0	14	1400	70	2 h	12	$7.3*10^{-4}$

The above tests were conducted on UO_2 rather than UCO. The conditions in the tests were not representative of conditions that would occur in the Xe-100 reactor, in that unlimited air was supplied to

the furnace in which the tests were conducted. Postulated accident conditions for modular HTGRs that involve a breach of the reactor helium pressure boundary and the ingress of oxidants are expected to result in an air/helium mixture of the order of a few percent air to which the reactor core is exposed.

It has been recognized that the AGR Program should include PIH tests with UCO fuel in oxidizing atmospheres that are more representative of conditions that could occur in the reactor than the KORA furnace tests described above. INL is currently designing an oxidants furnace for performing such tests as part of the AGR Program [Stempien 2016] [116].

5.5 FUEL PERFORMANCE MODEL DEVELOPMENT, VALIDATION, AND USE

All of the major international HTGR technology-development programs, including the U.S. HTGR program, have included extensive efforts to model the performance of TRISO-coated particle fuel. These development efforts have addressed the structural, thermal, and chemical processes that can lead to particle failures, as described in Section 5.1.4. Fission-product release from the particles and fuel elements, both prismatic and pebble, have also been modeled. The models that have been developed have, as discussed in [Loza 2021] [3], been used in calculations of mechanistic source terms that are conservative with respect to measured values. This section discusses the development and validation of fuel performance models for both spherical and prismatic fuel elements and the approach that will be taken to develop and validate the fuel performance model for the Xe-100. More detailed information on the latter, including a description of all of the models used in calculation of Xe-100 mechanistic source terms, is contained in [Loza 2021] [3].

5.5.1 Fuel Performance Models for Spherical Fuel Elements

The development of analytical models to predict the performance of TRISO fuel particles in spherical fuel elements during normal operation and postulated accidents began in the early 1970s as part of the German HTR program and was further advanced by the PBMR Project which built upon the German modeling efforts. In addition, as these models were developed, aggressive efforts to validate them by comparison of code predictions to actual performance data from an extensive series of German-sponsored irradiation tests in MTRs and from AVR surveillance data were made by the German program and later by the PBMR Project. The results have been documented in various review reports, including three IAEA Technical Documents [TEC-978 1997] [69], [TEC-1645 2010] [70], and [TEC-1674 2012] [23]].

The modules of X-energy's mechanistic source term code suite XSTERM, under development, for predicting TRISO fuel performance (XFP) and FP release from the fuel spheres (XGAS and XSOL) are described in [Loza 2021] [3]. These X-energy codes will improve upon the codes originally developed the German program and later utilized by the PBMR Project (PANAMA, NOBLEG, GETTER, etc.). The improvements will be to incorporate the models that fit the AGR UCO test data, to use more accurate material data as a result of AGR tests, and to use better physics in models as learned from the PARFUME code and/or available international publications. Since the X-energy codes will still use earlier codes as a reference, the earlier efforts by the Germans and PBMR to validate these codes are relevant to the new X-energy codes as well. Several examples of these validation efforts, entailing comparisons of calculated and measured radionuclide release from several individual test spheres, are presented in [Loza 2021] [3] and summarized in [Hanson 2016] [117].

These comparisons of measured and calculated data indicate that the source term codes historically used to evaluate source terms in pebble bed reactors can predict RN transport and release conservatively and within the design factors commonly accepted for comparisons of calculations and measurements in pebble bed HTGRs, as described in Section 4.

5.5.2 Fuel Performance Models for Prismatic Fuel Elements

Models for predicting TRISO fuel performance and FP release from prismatic fuel elements were under development in U.S. beginning in the late 1960s and mostly at General Atomics (GA). The most comprehensive compendium of these GA models is [Martin 1993] [73]. As on the German and PBMR programs, numerous efforts were made by the earlier U.S. HTGR programs to validate these models by comparison of code predictions with experimental data from irradiation tests in MTRs and with reactor surveillance data [e.g., Hanson 2004] [118]. Examples of the results of such efforts, including comparisons of calculated and measured gaseous and condensable metallic radionuclide release from the core of the FSV Nuclear Generating Station, are presented in [Loza 2021] [3]. As is the case for the comparisons referred to in Section 5.5, the source term codes historically used to evaluate source terms in prismatic block reactors also can predict RN transport and release conservatively and within the commonly accepted design factors described in Section 4.

The fuel performance models developed in the 1970s through 1990s by earlier U.S. HTGR programs are largely fits of experimental data to phenomenological models. Along with developing better material-properties data, the fuel performance modeling effort under the AGR Fuel Program is focused on developing the Particle Fuel Code (PARFUME) at INL to predict fuel failure based on a first-principles understanding of the phenomena that influence fuel performance. Considerable progress with respect to development of the PARFUME code has been made to date, but more both more reliable models and/or more accurate material data (especially the diffusivity) are needed, given the large discrepancies between PARFUME predictions and AGR test results [e.g., Collin 2015b [121], Collin 2016 [122]].

Benchmarking of international fuel performance codes for both normal reactor operating conditions and accident conditions was conducted under an IAEA cooperative research program (CRP-6) [TEC-1674 2012] [23]. Additional fuel performance code benchmarking is expected under the Gen IV VHTR Fuel and Fuel Cycle Program Management Board, based on the behavior of the current generation of TRISO fuel in planned irradiation and safety tests. Pretest predictions and after-test calculations will be performed for

each irradiation test and some of the safety tests. It is expected that the various irradiations and safety tests planned by the NGNP/AGR Fuel Program will provide much of the data needed to improve the fuel-performance models.

5.5.3 Use of Fuel Performance Models in Calculation of Xe-100 Mechanistic Source Terms

Several analytical tools are used to calculate fuel performance and FP generation, transport, and release to the environment. The purpose of this section is to provide a brief discussion of the fuel performance codes under development for the design and analysis of the Xe-100. The codes of direct interest here are those that calculate the performance of the UCO TRISO fuel particle. The results of these calculations are used as input for the various fission product transport codes, as described in [Loza 2021] [3].

Many calculational models exist to estimate TRISO fuel performance during irradiation and postulated accidents. From a licensing perspective, the methods used to predict fuel behavior to determine the source term need to be validated by experimental data or benchmarked against other validated codes. Modern coated particles experience very low failure rates during irradiation, and the failed fuel fraction is typically dominated by the small fraction of particles with manufacturing defects, which is measured during fuel fabrication. The codes to be used to predict source terms for the Xe-100 are summarized in [Olivier 2016] [123] and in [Loza 2021] [3].

The fuel performance code XFP is to be used for determining the failed particle probabilities from the identified fuel failure mechanisms discussed in Section 5.1.4. The individual failure probabilities are combined to determine a fuel failure fraction for the given operating conditions. XFP can calculate the failure fractions in each core node based on the operating conditions such as irradiation temperature, transient temperature, and neutron fluence. These failure fractions are transferred to the XSOL and XGAS codes as inputs into radionuclide transport and release calculations.

Fuel failure fractions in the reactor core vary in space and time as do the fractional releases from heavy metal (HM) contamination and failed particles. FP transport behavior in the reactor core and around the primary circuit varies by species and with temperature and is affected by the materials used in the core and the primary cooling system. Consequently, full-core computer codes and models of the entire primary circuit are needed to track these effects.

Two different approaches will be applied for the X-energy fuel performance calculations. The initial effort is to implement all the physical failure mechanisms discussed in Section 5.1.4 in XFP. The XFP fuel performance model will, accordingly, be more advanced than the German fuel performance code PANAMA, which only includes the pressure vessel failure model. It is anticipated that the X-energy fuel performance code will be similar to the state-of-the-art US fuel performance code PARFUME under development by INL. Once completed, the XFP code will be benchmarked against PANAMA and PARFUME to determine the level of agreement and isolate the main sources of disagreement between the codes. One candidate set of problems for code comparison are the benchmarks defined for IAEA CRP-6 [TEC-1674 2012] [23].

As a part of fuel performance code development, the second approach will include phenomenological models normalized to test data, especially the AGR test data. X-energy works with INL and ORNL to

develop UCO particle fuel performance models and correlations for use in the X-energy codes by acquiring and correlating relevant data.

The demonstrated capability to predict with sufficient accuracy full core fuel performance and FP release under both normal operating conditions and accident conditions is fundamental to Xe-100 fuel qualification, MST development, and determination of required design margins. Verification and validation (V&V) of fuel performance codes are required for adequate evaluation model development and transient/accident analyses for licensing. [[

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5.6 IMPLICATIONS FOR DEVELOPMENT AND QUALIFICATION OF UCO TRISO IN SPHERES

There is a robust national and international data base that provides the technical foundation for the development and qualification of the reference Xe-100 fuel design of UCO TRISO in spherical elements. As shown in Table 17:, the Xe-100 fuel design is a natural evolution of the TRISO fuel designs first produced by the German HTR program some four decades ago. The Germans have produced and irradiated large quantities of high-quality LEU UO_2 TRISO-coated particles for AVR reloads (Table 5) as well as HEU UCO TRISO-coated particles that were irradiated in spheres in AVR. The Chinese are currently producing large quantities of spheres of essentially the German design to fuel the two HTR-PM modules. Moreover, the AGR program has developed and is qualifying superior-performance LEU UCO TRISO-coated particles but in fuel compacts rather than spheres (Section 5.3.2). The task now is to demonstrate this UCO TRISO-coated particle will meet performance requirements in spheres as it does in compacts. The planned approach to accomplish that objective is described in Section 6.

[16] [[ ]]^{P}.

Table 17: Evolution of Spherical Fuel Element Designs

Attribute	AVR Reloads	HTR-Modul	PBMR	HTR-PM	Xe-100[a]
Kernel					
Enrichment (%)	10/17	8	7.8	8.5	15.5
Composition	UO_2	UO_2	UO_2	UO_2	UCO
Diameter (μm)	500	500	500	500	425
Kernel making Process	External gelation	External gelation	External gelation	Combined internal/external gelation	[[ ]]^P
TRISO Coatings					
Coating system	German	German	German design	German design and partial equipment	U.S.-AGR[b]
Sphere					
U loading (g/sphere)	10/6	7	9	7	7
# of particles/sphere	16,400/10,000	14,500	14,400	12,000	[ ]]^P
Matrix Material	A3-3	A3-3	A3-3	A3-3	US A3-3[c]
Fuel-free zone (mm)	0.5	0.5	0.5	0.4	0.5
Quality Requirements (ambiguity about confidence levels; Xe-100 value is mean value)					
HM contamination SiC defects Sum (burn/leach)	Not Specified Not Specified $\leq 6 \times 10^{-5}$	Not Specified Not Specified $\leq 6 \times 10^{-5}$	Not Specified Not Specified $\leq 6 \times 10^{-5}$	Not Specified Not Specified $\leq 6 \times 10^{-5}$	$\leq 1 \times 10^{-5}$ $\leq 5 \times 10^{-5}$ $\leq 6 \times 10^{-5}$
References	[Röser 1987] [118]	[HOBEG 1989] [60]	[NGNP FQ WP 2010] [4]	[Zhou 2013] [10]	[Marshall 2016] [106] [HOBEG 1989] [60]

[a]Xe-100 reference fuel is AGR-5/6/7 TRISO [Marshall 2016] [106] in spheres [HOBEG 1989] [60] with US A3-3 matrix.
[b] [[ ]]^P
[c]U.S. equivalent of German A3-3 matrix.

6 XE-100 FUEL QUALIFICATION PROGRAM

This section addresses the qualification of the fuel for the Xe-100 design. Section 6.1 addresses the basic considerations involved in fuel qualification, and Section 6.2 describes the fuel qualification approach adopted for the Xe-100 design.

6.1 BASIC CONSIDERATIONS

Qualification of the fuel for a modular HTGR requires fabrication in compliance with comprehensive product and process specifications to produce fuel demonstrating the required performance under normal operation and accident conditions. Common elements of a fuel qualification program include:

- Establishment of fuel product, process, equipment, and feedstock specifications;
- Implementation of a fuel fabrication process capable of consistently and reliably meeting the specifications at the required scale;
- Implementation of statistical QC/QA procedures to demonstrate that the product meets these specifications;
- Irradiation of statistically sufficient quantities of fuel with monitoring of in-pile performance and PIE to demonstrate that normal operation performance requirements are met; and
- Safety testing of statistically sufficient quantities of irradiated fuel to demonstrate that accident condition performance requirements are met.

For spherical fuel elements there are additional fuel qualification tasks associated with their various physical properties. For example, the spheres are tested to confirm that their structural integrity is maintained through repeated cycling through the core and within the fuel handling and storage systems. The methods for confirming structural integrity and other important physical properties are briefly summarized in Appendix C. FUEL SPHERE QUALITY TESTING

6.2 FUEL QUALIFICATION PROGRAM FOR THE XE-100 DESIGN

6.2.1 Introduction and Background

As described in Section 2, X-energy is developing the Xe-100, a 200 MWt pebble-bed modular HTGR, as an advanced nuclear power source. Traditionally, pebble-bed modular HTGRs have used LEU UO_2 TRISO fuel particles in spherical fuel elements. However, X-energy has chosen to adopt the LEU UCO TRISO-coated particle being developed and qualified by the AGR Fuel Development and Qualification Program as their reference particle to take advantage of the higher temperature and higher burnup capabilities of UCO [e.g., Petti 2014] [63]. This choice leverages the successful results of the AGR Program, which have been presented to both the NRC staff and the ACRS and were obtained under a DOE-approved QA program that complies with NQA-1 requirements.

The Xe-100 design needs a qualified fuel and validated methods for predicting fuel performance and source terms to facilitate successful plant design and licensing. As described in Section 5.3.2, the AGR

Program is qualifying the reference UCO particle but in fuel compacts that are characteristic of prismatic HTGR cores rather than in fuel spheres. X-energy anticipates that the AGR UCO compact data and particle performance will apply to the same UCO particles manufactured into spheres. Based on the German experience with HEU UCO fuel particles in spheres (Section 5.6), it is likely that the LEU UCO TRISO-coated particle will perform in spheres as it has been observed to perform in fuel compacts even though the process conditions for making spheres are somewhat different (and to higher pressures) than for making fuel compacts. In particular, the pressures for forming spheres are considerably higher than for making compacts (typically, 300 MPa for sphere-pressing compared to less than 20 MPa for compacting. This difference is, however, somewhat offset by the lower packing fraction of the particles in the sphere volume (~10%) compared with that of particles in the cylindrical compact volume (~35-40%). The lower packing fraction results in less likelihood of contact between the outer surfaces of particles. The successful fabrication and irradiation of thousands of fuel spheres in the AVR reactor implies that TRISO-coated fuel particles in spheres can be successfully fabricated in the U.S. using existing, proven methods.

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6.2.2 Overview of the Xe-100 Fuel Development and Qualification Approach

The X-energy fuel qualification program is succinctly summarized in Table 18.

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Table 18. [[ ]]P

Name (Fuel Source/ Reactor)	Test Objectives	Test Description
[[ ]]P	[[ ]]P	[[ ]]P
[[ ]]P	[[ ]]P	[[ ]]P

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6.2.3 Detailed Description of the Xe-series Fuel Qualification Tests

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[17]In fact, the consistent performance behavior of German LEU UO_2 TRISO fuel irradiated in AVR vs. in water-cooled MTRs indicates that the spectral differences are unimportant (Section 5.2.1.2).

Table 19. [[ ]]^{P}

Capsule	Test Objective/Description	Test Fuel			ATR Test Train	Irradiation (Peak Conditions)			
		Production Equipment	U-235 Enrich. (%)	Kernel Diameter (µm)		Burnup (% FIMA)	Fast Fluence (n/m^2)	TAVA[1] (°C)	Duration (EFPD)
AGR-1	**Shakedown Test/Early Fuel** Qualify the new multi-capsule test train. Test included fuel compacts made from UCO fuel particles coated in a 2-inch laboratory scale coater at ORNL. A baseline fuel particle composite and three variant fuel particle composites were tested (two lPyC and one SiC variant).	Lab-scale	19.7	350	6 capsules	19.6	4.3×10^{25}	1136	620
AGR-2[2]	**Performance Test Fuel** Test fuel included fuel compacts containing UCO particles made in 6" coater and UO_2 particles made by B&W, AREVA, and PBMR in different size coaters. One capsule of UCO fuel was designed to operate with a TAVA of 1400°C as a performance margin test.	6" coater	14.0	425	6 capsules	13.2	3.5×10^{25}	1296	559

Capsule	Test Objective/Description	Test Fuel			ATR Test Train	Irradiation (Peak Conditions)			
		Production Equipment	U-235 Enrich. (%)	Kernel Diameter (µm)		Burnup (% FIMA)	Fast Fluence (n/m^2)	TAVA[1] (°C)	Duration (EFPD)
AGR-3/4 (Data should be sufficient to meet Xe-100 FP release DDNs. No need to repeat with pebble fuel.)	**Fission Product Transport** Test fuel included compacts of LEU UCO particles seeded with designed-to-fail (DTF) fuel LEU UCO particles to provide a well-defined FP source. Fuel test element utilized a concentric-ring design to provide a 1-D geometry to facilitate derivation of effective diffusivities for fission metals in matrix and graphite.	Lab-scale	19.7	350	12 capsules	15.3	5.3×10^{25}	1299	369

Capsule	Test Objective/Description	Test Fuel			ATR Test Train	Irradiation (Peak Conditions)			
		Production Equipment	U-235 Enrich. (%)	Kernel Diameter (µm)		Burnup (% FIMA)	Fast Fluence (n/m^2)	TAVA[1] (°C)	Duration (EFPD)
AGR-5/6/7[3]	**Fuel Qualification** Two tests to be irradiated in the same test train. Test fuel is being made in pilot plant-scale equipment, including 6" coater, using optimized process conditions based upon AGR-1 and -2 test results. **AGR-5/6** is a qualification test with a statistically significant particle population irradiated over a range of temperatures to demonstrate fuel performance capabilities at required confidence level to meet reactor requirements. **AGR-7** is a margin test at ~1500°C to quantify the fuel's high-temperature capabilities and to demonstrate the absence of a performance "cliff" to well beyond the fuel service envelope.	Pilot plant-scale	15.5	425	4 capsules	(>18)	(<7.5 x 10^{25})	(1350 ± 50)[4]	(500)[4]

Capsule	Test Objective/Description	Test Fuel			ATR Test Train	Irradiation (Peak Conditions)			
		Production Equipment	U-235 Enrich. (%)	Kernel Diameter (µm)		Burnup (% FIMA)	Fast Fluence (n/m^2)	TAVA[1] (°C)	Duration (EFPD)
[[ ]]^P	[[ ]]^P	[[ ]]^P	[[ ]]^P	[[ ]]^P	[[ ]]^P	[[ ]]^P	[[ ]]^P	[[ ]]^P	[[ ]]^P

Capsule	Test Objective/Description	Test Fuel			ATR Test Train	Irradiation (Peak Conditions)			
		Production Equipment	U-235 Enrich. (%)	Kernel Diameter (µm)		Burnup (% FIMA)	Fast Fluence (n/m^2)	TAVA[1] (°C)	Duration (EFPD)
[ ]P	[ ]] P	[ ]] P	[[ ]]P	[[ ]]P	[ ]] P	[[ ]] P	[[ ]] P	[[ ]] P	[[ ]] P
[ ]] P	[ ]] P	[ ]] P	[[ ]]P	[[ ]]P	[ ]] P	[[ ]] P	[[ ]] P	[[ ]] P	[[ ]] P

[1]TAVA = Time-average, (capsule) volume-average temperature.
[2]Three capsules with LEU UCO fuel and three capsules with LEU UO_2 fuel.
[3]AGR-5/6 and AGR-7 to be irradiated in a single test train, but they have different test objectives and service conditions.
[4]AGR-5/6/7 Duration was cut short due to experiencing unexpected/abnormal results.

Table 20. [[ ]]P

Test	Test Description	Capsule	Irradiation TAVA (°C)	PIE	Post-irradiation Heating (PIH)[1]		PIH PIE
					(°C)	Atmosphere	
[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P
		[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P
		[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P
		[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P
		[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P
		[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P
		[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P
		[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P
		[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P
		[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P
		[[ ]]P	[[ ]]P	[[ ]]P			
		[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P
		[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P

Test	Test Description	Capsule	Irradiation TAVA (°C)	PIE	Post-irradiation Heating (PIH)[1]		PIH PIE
					(°C)	Atmosphere	
		[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P
		[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P	[[ ]]P
[ ]P	[ ]]P	[ ]]P	[[ ]]P	[ ]]P			
		[ ]]P	[[ ]]P	[ ]]P	[ ]]P	[ ]]P	[ ]]P
		[ ]]P	[[ ]]P	[ ]]P	[ ]]P	[ ]]P	[ ]]P
		[ ]]P	[[ ]]P	[ ]]P	[[ ]]P	[ ]]P	[ ]]P
[ ]P	[ ]]P	[ ]]P	[[ ]]P	[ ]]P			
		[ ]]P	[[ ]]P	[ ]]P	[ ]]P	[ ]]P	[ ]]P
		[ ]]P	[[ ]]P	[ ]]P	[ ]]P	[ ]]P	[ ]]P
		[ ]]P	[[ ]]P	[ ]]P	[[ ]]P	[ ]]P	[ ]]P

[1]Conditions of PIH tests subject to change depending upon results of AGR PIH tests (e.g., tests in He/H_2O could be done at 1300°C rather 1500°C).

Table 21. Test Conditions for the AGR Irradiation Tests of UCO in Compacts

Capsule	Test Fuel			ATR Test Train	Irradiation (Peak Conditions)					INL Reference
	Production Equipment	U-235 Enrich. (%)	Kernel Diameter (µm)		Acceleration Factor[1]	Burnup (% FIMA)	Fast Fluence (n/m^2)	TAVA (°C)	Duration (EFPD)	
AGR-1	Lab-scale	19.7	350	6 capsules	1.7	19.6	4.3×10^{25}	1087	620	INL/EXT-10-18097
AGR-2[2]	6" coater	14.0	425	6 capsules	1.9	13.2	3.5×10^{25}	1296	559	INL/EXT-14-32277
AGR-3/4	Lab-scale	19.7	350	12 capsules	2.9	15.3	5.3×10^{25}	1276	369	INL/EXT-15-35550
AGR-5/6[3]	Pilot scale	15.5	425	4 capsules	2.1	(>18)	($<7.5 \times 10^{25}$)	(1350 ± 50)	(500)	INL SPC-1749
AGR-7[3]				1 capsule	(2.1)	(>18)	($<7.5 \times 10^{25}$)	(1500 ± 50)	(500)	

[1]Acceleration Factor (AF) calculated by dividing Fuel Residence Time in SC-MHR (1060 EFPD) by the Test Duration.
[2]Three capsules with LEU UCO fuel and three capsules with LEU UO_2 fuel; AGR-2/2 is margin test of UCO fuel.
[3]AGR-5/6 (4 capsules) and AGR-7 (1 capsule) to be irradiated in a single test train, but tests have different test objectives and service conditions.

6.2.4 Post-irradiation Examination and Safety Testing Protocol

The protocol for post-irradiation characterization of the test fuel [e.g., AGR TDP 2016] [18] is shown in a simplified flow chart in Figure 38. A non-destructive PIE is performed for all test articles recovered from the test train. After that, the test articles can be designated for post-irradiation heating (PIH) tests or for destructive PIE, which yields far more information on the effects of irradiation on the performance of the fuel than does the non-destructive PIE. In the case of a fuel-compact test, which yields a large number of test specimens, some compacts are simply archived after non-destructive PIE. In the case of a fuel sphere test, which yields relatively fewer test specimens, all of the irradiated spheres would either undergo destructive PIE or would be heated. After the heating tests, the test specimens typically also undergo destructive PIE.

Ideally, some or all of the fuel compacts and fuel spheres should undergo a short-term re-irradiation to produce inventories of radiologically important, short-lived fission gases, especially 8-day I-131 and 5.3-day Xe-133, prior to heating. However, the AGR Program has yet to establish a capability to re-irradiate compacts because of budget constraints. As an alternative, the current plan is to re-irradiate small batches of loose particles that have been recovered from deconsolidated fuel compacts and have been mechanically failed. This approach can also be followed for spheres if the capability to re-irradiate fuel compacts and spheres is not developed.

The standard AGR PIE protocol was developed to characterize TRISO-coated particles in fuel compacts. Some modifications of the procedures will be required for the characterization of irradiated fuel spheres (e.g., deconsolidation, leach/burn/leach, etc.) because of their large size. Fortunately, the requisite procedures for spheres have been largely developed by decades of PIE experience with fuel spheres, especially in Germany (e.g., Section 8 of [TEC-1674 2012] [23]).

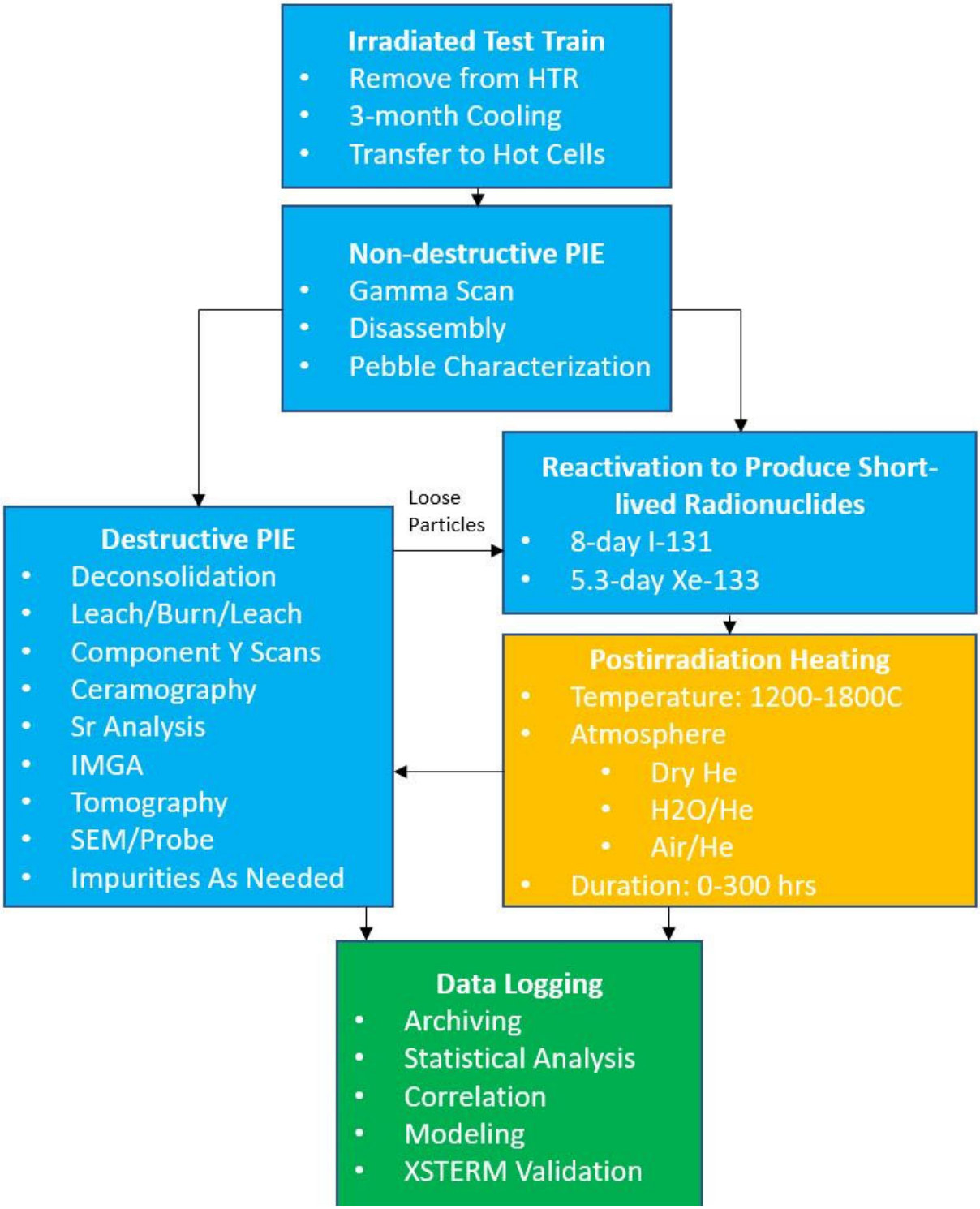

Figure 38: Simplified Flowchart of the PIE Protocol

6.2.5 Application of the AGR Experience to Testing of UCO Spheres

Overall, the various AGR test trains have performed superbly, with the only significant problems being premature failures of very small-diameter thermocouples in AGR-1 and AGR-2 and the leakage and cross-talk of the purge flows in AGR-2. The latter problem was a result of removing the experiment during high power runs, which damaged the gas lines. This problem was probably more significant since the He/Ne mix of the purge gas was used to control the temperatures in the individual capsules. [[

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Although the AGR-1 and 2 irradiations do not match the design of the Xe-100 TRISO in pebbles, the data is still valuable in establishing the operating envelope for TRISO fuel in general. EPRI recognized the importance of these two tests and summarized them in a Topical Report to the NRC [EPRI-AR-1] [1]. The TR was approved [EPRI-AR-1 FSER] [7] and the NRC issued their August 2020 safety evaluation on the basis of three key conclusions:

1. The performance of the UCO TRISO-coated fuel particles over the tested range of conditions provides a foundational basis for use of these particle designs in the fuel elements of TRISO-fueled HTR designs including the x-energy design.

2. Despite the range of properties and fabrications, the kernels and coatings of the UCO TRISO-coated fuel particles tested exhibited remarkably similar excellent irradiation and accident safety performance, with variations in key characteristics reflected in measured particle layer properties. UCO TRISO-coated fuel particles that satisfy the parameter envelope defined by these measured particle layer properties can be relied upon to provide satisfactory performance.

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

Results indicate that the form of the fuel element (i.e. cylindrical compact or spherical pebble) does not impact TRISO particle performance, given that the particle meets the TRISO particle manufacturing specification. Again, this makes the AGR-1 and 2 data, which was acquired from cylindrical compacts, applicable to TRISO in spherical pebbles.

6.2.6 [[ ]]^{P}

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Figure 39. [[ ]]^{P}

[[ ]]^{P}

Figure 40. [[ ]]^{P}

[[ ]]^{P}

Figure 41. [[ ]]^{P}

7 NRC TOPICAL REPORT REVIEW OBJECTIVES

The information in this report is intended to serve a basis for interaction with the NRC and other regulatory agency staffs and provide the methodology by which TRISO-X coated-particle fuel will be qualified for use in the Xe-100 reactor. This report focuses on the use of the AGR program UCO data base along with the national and international TRISO fuel-performance data base **[[**]]^{P} to establish the bases of fuel performance under the intended design conditions. X-energy requests the NRC staff review and approve the following conclusions from this report:

- The fuel design and performance requirements in Section 4 are adequate for establishing an acceptable design basis to support the licensing of the Xe-100 reactor.
- Plans established in Section 6 for qualification of the UCO TRISO-coated particles in spheres are generally acceptable. These include:
  - Utilization of the AGR UCO compact data for normal operation and transient/accident heat-up conditions,
  - Performance of confirmatory irradiation, post-irradiation examination, and safety tests on spherical fuel elements manufactured on a pilot line under an NQA-1 Quality Assurance Program to demonstrate that UCO fuel performance in spheres meets requirements,
  - **[[**]]^{P}
  - **[[**]]^{P}

X-energy supports further dialogue with the NRC staff during the review of this topical report, and expects that requests for additional information will identify questions regarding research, development, or testing needed to demonstrate adequate Xe-100 fuel performance.

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APPENDIX A. NRC ASSESSMENT OF NGNP WHITE PAPERS: FOLLOW UP ITEMS FOR FUEL QUALIFICATION

As discussed in Section 3, the NRC staff assessment report that was prepared to document the staff's review of the NGNP Fuel Qualification and Mechanistic Source Term white papers documented several "items for follow up" that the NRC would continue to examine in future licensing interactions for modular HTGRs. The staff combined their review of the two white papers in one assessment report because the mechanistic source term is very strongly dependent on the fuel performance issues addressed in the Fuel Qualification White Paper. A preliminary draft assessment was issued by a staff NGNP working group in February 2012 [NRC 2012a] [42]. A final assessment was issued in July 2014, following more than two years of additional focused discussions in public meetings between the NRC staff and DOE/INL and meetings with the ACRS in January, April, and May, 2013 [NRC 2014] [43].

The follow up items from the 2012 draft NRC assessment report that are directly related to fuel qualification are summarized in Table A-1. Each item in the table is classified as being predominantly a fuel qualification item. Both classifications of items are presented in the table to provide a complete picture of the staff positions. The section and page numbers of each item from the February 2012 draft assessment report are provided, as well as a description of the item. Following the release of the draft assessments, additional interaction took place between the NGNP Project and the NRC staff. In July 2014, the NRC staff released its final assessment of the Fuel Qualification and Mechanistic Source Term white papers [NRC 2014] [43]. The updated assessment items of July 2014 are also presented in Table A-1 In some cases, items were satisfactorily addressed by the NGNP staff in the subsequent meetings and were noted by the staff as resolved in the 2014 final assessment report. This is indicated as appropriate in Table A-1 In other cases the follow up items were unchanged or expanded upon to provide more information on the staff position which is also shown in the table. The table also provides the accession numbers from the NRC public document database (ADAMS) for the draft and final assessment reports.

Most of these items for follow up will be addressed by X-energy as the design of the Xe-100 progresses or as the technology development activities described in Section 6 are completed.

Table A-1. NRC NGNP Assessment Follow Up Items

Item No.	Topic	Draft Assess Sec/Pg	February 2012 Draft Assessment (ML120240669)	July 2014 Final Assessment (ML14174A845)
1	Fuel	3.2.1/ 12	**Fuel Service Conditions for Normal Operations:** Currently, like the NGNP design itself, the fuel design service conditions for NGNP normal operations are not yet finalized and are therefore considered items for follow up.	Currently, like the NGNP design itself, the fuel design service conditions for NGNP normal operations are not yet finalized and will have to be further specified by a future applicant. The normal operating fuel service conditions addressed in the NGNP/AGR Fuel Program's normal operation irradiation tests are presently based on what DOE/INL states to be a conservative assessment of the best available code predictions of fuel operating conditions in preliminary designs of an NGNP prismatic block core. When NGNP normal fuel service conditions have been finalized, it will be necessary to show how well they are addressed by those tested in the NGNP/AGR Fuel Program.
2	Fuel	3.2.1/ 13	**Additional Fuel Operating Condition Parameters:** Fuel service conditions for normal operations should be supplemented with parameters that address maximum plutonium burnup (fissions from bred plutonium) and maximum time-at-temperature.	Item unchanged from Draft Assessment Report.

Item No.	Topic	Draft Assess Sec/Pg	February 2012 Draft Assessment (ML120240669)	July 2014 Final Assessment (ML14174A845)
3	Fuel	3.2.1/ 13	**Parameter Path Dependence:** Information on how fuel service conditions vary with location and operating time may be needed to address "parameter path dependence". Accelerated testing addresses combinations of high fluence, burnup, and temperature. Other combinations, such as high fluence with moderate bumup or moderate fluence with high burnup or low temperature with high fluence may need to be considered.	In follow-on discussions, DOE/INL stated that it has not seen evidence of parameter path dependence for normal fuel operating conditions but acknowledged the need to further evaluate this issue based on NGNP core design information yet to be established (ML12132A467). The staff notes that, once a reference NGNP core design has been sufficiently established, calculated end-of-cycle core maps of fuel irradiation parameter combinations (i.e., fuel temperature, bumup, fluence) should be provided and compared for coverage against the tested irradiation parameter combinations realized in the NGNP/AGR Fuel Program. The potential significance of any "path dependence" coverage gaps thus found should then be analyzed and evaluated using validated phenomenological models of TRISO fuel performance under operating conditions and accident conditions.
4	Fuel	3.2.1/ 13	**Operating Condition Uncertainties and Anomalies:** Issues of uncertainties in HTGR core analysis and core monitoring can be addressed only in small part by analytical means and separate-effects validation testing. Resolution of these issues will likely necessitate verification of initial and evolving NGNP normal fuel operating conditions and performance through special operational monitoring, testing, surveillance, and inspection programs for the NGNP prototype.	In related RAI responses, DOE/INL has acknowledged this issue as one that should be addressed as detailed design information is developed for any future NGNP application. The staff agrees but also notes that bringing earlier attention to this issue could benefit the timely development and qualification of advanced sensor systems for NGNP prototype monitoring, surveillance, and testing, and that basic technical requirements for such sensor systems may prove largely generic to all modular HTGR design variants.
5	Fuel	3.2.2/ 14	**Fuel Service Conditions for Accidents:** The fuel design service conditions for NGNP accidents, like those for NGNP normal operations, have not yet been finalized and are considered items for follow up. When finalized, it will thus be necessary to show that they have been adequately addressed by the fuel accident conditions tested in the NGNP/AGR Fuel Program.	Item unchanged from Draft Assessment Report. (It was the position of DOE/INL that this item cannot be completely addressed until the NGNP design is further advanced.)

Item No.	Topic	Draft Assess Sec/Pg	February 2012 Draft Assessment (ML120240669)	July 2014 Final Assessment (ML14174A845)
6	Fuel	3.2.2/ 15	**Reactivity excursions, air and moisture ingress events:** Fuel accident conditions and performance requirements for (a) reactivity excursions, (b) air ingress events, and (c) moisture ingress events are considered major items for follow up.	Fuel accident conditions and performance requirements for (a) reactivity excursion events, (b) operating core "hot spot" events, (c) air ingress events, and (d) moisture ingress events should be further addressed as relevant NGNP design and analysis details are established. (It was the position of DOE/INL that this item cannot be completely addressed until the NGNP design is further advanced.)

Item No.	Topic	Draft Assess Sec/Pg	February 2012 Draft Assessment (ML120240669)	July 2014 Final Assessment (ML14174A845)
7	Fuel	3.2.3/ 16	**Fuel Performance Terminology:** The Project should establish explicit definitions with descriptive terms like defective, failed, and functionally-failed relative to fuel particles and individual coating layers and explain how fuel performance and radionuclide transport and release are considered and modeled in each case.	In describing these concerns in the initial assessment report, the NRC staff suggested that terms like "defective," "failed," and "functionally-failed" should be used to describe fuel particles in relation to the condition of individual coating layers and explain how fuel performance and radionuclide transport and release are considered and modeled in each case. DOE/INL subsequently responded in a public meeting by reporting that, in addition to the terms "intact" and "failed" used in the FQ white paper, future NGNP submittals would use the term "functionally degraded" to describe, for example, a fuel particle with intact PyC layers that retain fission gases and a defective or degraded SiC layer that allows the release of additional fission metals (e.g., cesium) (ML12132A467). The staff agrees that using this additional descriptive term can help bring necessary clarity to the evaluation and modeling of TRISO fuel performance. (DOE/INL presented the following definitions in a public meeting on 4/17/12: Intact Particle: A particle with all coatings structurally intact. Functionally Degraded Particle: A particle with one or more coatings degraded such that additional metallic fission products are released. Failed Particle: A particle with an open pathway from the kernel to the outer surface of the particle. "Defective particle" describes an as-manufactured deficiency. DOE/INL stated that models of coated particle fuel performance and radionuclide release take into account all fuel particle conditions that affect particle performance and radionuclide release.)

Item No.	Topic	Draft Assess Sec/Pg	February 2012 Draft Assessment (ML120240669)	July 2014 Final Assessment (ML14174A845)
8	Fuel	3.3/ 17	**Fuel Design and Product Specifications:** The ultimate adequacy of these specifications will depend on the outcome of the AGR-3/4 fuel fission product transport data development tests, the AGR-5/6 fuel qualification tests and the AGR-7/8 fuel fission product transport code validation tests. The outcome of these tests will indicate the NGNP safety analysis codes and methods uncertainties and/or biases that must be accommodated in the NGNP safety analysis. These tests and their effects on the fuel product specifications are follow-up items.	Item unchanged from Draft Assessment Report.
9	Fuel	3.4/ 19	**Fuel Fabrication Variability:** The variability on fuel attributes for a production facility may be different from the fuel attribute variability for fuel made with a single line. Simulating the large fuel fabrication facility variability might be achieved by mixing several runs (i.e., batches, lots) from the single line. Whether and how the Project plans to address differences in the variability of product attributes between fuel fabricated for AGR-5/6 with a single line of "production-scale" equipment and fuel fabricated for the NGNP prototype with multiple lines of equipment in a large fuel fabrication facility is a follow-up item.	Item not included in Final Assessment Report.

Item No.	Topic	Draft Assess Sec/Pg	February 2012 Draft Assessment (ML120240669)	July 2014 Final Assessment (ML14174A845)
10	Fuel	3.5/ 20	**Fuel Characterization and Quality Control:** The fuel characterization methods and fuel fabrication statistical quality control procedures used for the NGNP/AGR fuel qualification program and NGNP reactor production fuel fabrication facility are follow-up items.	In follow-on discussions, the NRC staff and DOE/INL agreed that completion of these items will be handled by the NGNP reactor and fuel vendors and the NGNP applicant. DOE/INL confirmed in a public meeting that the statistics of fuel characterization demand that the specifications be met with margin to keep the size of the sampled population of fuel particles manageable during production. Going forward, issues of concern to the NRC staff would include the extent to which NRC review and monitoring of fuel fabrication process parameters is needed and the extent to which fuel product characterization methods and procedures at the fabrication facility may vary from those used for the NGNP/AGR Fuel Program. DOE/INL offered to share any updated information related to these issues as it becomes available (ML12132A467).

Item No.	Topic	Draft Assess Sec/Pg	February 2012 Draft Assessment (ML120240669)	July 2014 Final Assessment (ML14174A845)
11	Fuel	3.6.1/ 21-22	**Adequacy of Accelerated Irradiation Testing:** Lack of real time testing of fuel in an HTGR environment raises concerns regarding the adequacy with which phenomena that depend on plutonium burnup and time at temperature are characterized. Proof testing in the first modular HTGR can address these issues. The inclusion of a suitably designed post-irradiation fuel inspection and testing program for the NGNP prototype can provide the important confirmatory fuel performance data for the UCO TRISO coated fuel particle design. Validation of the NGNP fuel performance/fission product transport models and codes may require an effort completely independent from data gathering inspections. Independent code modeling predictions, followed by an independent evaluation of reference fuel performance under normal operating and accident condition simulation carried out on irradiated fuel from the NGNP prototype, appear to be feasible. Satisfactory completion of such a prescribed phase in a post-irradiation fuel inspection and testing program achieved with irradiated fuel from the NGNP prototype is considered a final confirmatory step in the NGNP UCO fuel qualification program. The working group considers this issue a major follow-up item.	Item unchanged from the Draft Assessment Report except for the addition of the following: The NRC staff believes satisfactory completion of a post-irradiation fuel inspection and testing program achieved with irradiated fuel from the NGNP prototype is necessary to verify and supplement the technical basis for NGNP fuel qualification and mechanistic source terms code validation. (It was the position of DOE/INL that this item will be addressed by the COL applicant as part of the COL review.)
12	Fuel	3.6.2/ 24	**Prototypical Irradiation Testing Neutron Spectrum:** The NGNP approach to increasing plutonium burnup in the AGR irradiation tests relies solely on using neutron absorbers in the test rig to effectively harden the thermal spectrum by reducing the neutron flux in the lower range of the ATR thermal energy spectrum. The working group presently views this approach as unlikely to adequately address plutonium burnup, time at operating temperature, and particularly palladium time-at-temperature, important parameters that should be considered in the irradiation testing of TRISO fuel.	Item unchanged from the Draft Assessment Report.

Item No.	Topic	Draft Assess Sec/Pg	February 2012 Draft Assessment (ML120240669)	July 2014 Final Assessment (ML14174A845)
13	Fuel	3.6.2/ 24	**Palladium, Silver, and Rare Earth Time at Temperature:** Plutonium burnup, time at operating temperature, and particularly palladium time-at-temperature are important parameters that should be considered in the irradiation testing of TRISO fuel. This issue area is considered a major item for follow up.	DOE/INL's RAI responses also included a requested summary of the current state of knowledge on how palladium, silver, and rare earth fission products can affect TRISO fuel performance. The staff further considered these issues during the follow-on assessment phase in light of supplemental information and observations provided by DOE/INL in TEV-1620 (ML12268A032). As summarized in TEV-1620, DOE/INL currently believes that emerging experimental evidence points to rare earths, palladium, and silver having little effect on NGNP TRISO fuel performance. Moreover, DOE/INL expects future results from the NGNP/AGR Fuel Program to further support this interpretation. The staff presently agrees with DOE/INL regarding the limited effects of rare earth fission products. However, on reviewing the information provided, the NRC staff notes that the evolving phenomenological understanding of how palladium and silver interact with TRISO fuel coatings is still very limited. Continued research is needed to support a compelling explanation for the sporadic cases of palladium attack on SiC that have been reported internationally in the TRISO fuel technical literature. The NRC staff therefore continues to view plutonium burnup, time at operating temperature, and particularly palladium time-at-temperature as important parameters that should be considered in the irradiation and accident testing of TRISO fuel. The staff intends to further evaluate this issue as continuing research efforts by DOE/INL and others yield new insights into how plutonium fission products interact with coating layers on the TRISO fuel particles now being developed and tested by the NGNP/AGR Fuel Program.

Item No.	Topic	Draft Assess Sec/Pg	February 2012 Draft Assessment (ML120240669)	July 2014 Final Assessment (ML14174A845)
14	Fuel	3.6.2/ 24	**Pre-irradiation Test Predictions:** The Project has not responded to the working group's RAI requesting pre-test predictions of the recently completed AGR-2 irradiation nor of any future AGR irradiations. The Project should freely share all test design and pre-test predictions of AGR irradiation conditions and irradiation fuel performance.	Item not included in Final Assessment Report
15	Fuel	3.6.3/ 24	**Access to Detailed ATR Information:** The working group has given consideration to performing independent NRC analyses of AGR test irradiation conditions and associated fuel burnup isotopics and would be willing to pursue arrangements for gaining access to the detailed ATR information that would be needed for doing so. This is an item for follow up.	In follow-on discussions, DOE/INL noted that data needed for thermal modeling of the irradiations could be made available whenever requested but that special arrangements with other agencies would be necessary for accessing the detailed information needed for nuclear analysis. (DOE/INL had noted that data needed to conduct an accurate independent nuclear analysis is classified. NRC would have to obtain the information from Naval Reactors.)

Item No.	Topic	Draft Assess Sec/Pg	February 2012 Draft Assessment (ML120240669)	July 2014 Final Assessment (ML14174A845)
16	Fuel	3.6.3/ 25	**AGR Fuel Irradiation Temperature Uncertainties**: Further assessment of AGR fuel irradiation temperature uncertainties and how they are affected by thermocouple failures is an item for follow up.	In response to a related RAI question on how such thermocouple failures are accounted for in evaluating irradiation temperatures and associated uncertainties, DOE/INL explained how thermocouples embedded in the graphite sample holders are used in conjunction with detailed analytical models to determine fuel temperatures in the AGR irradiation tests. As mentioned in the RAI responses, DOE/INL later provided more detailed information in INL-EXT-12-24761 and INL-EXT-12-25169, which are technical reports respectively detailing the analysis of thermocouple data and the quantification of temperature uncertainties for the AGR-1 irradiation test (ML12205A039). It bears noting that the latter report estimates standard deviations between 45 and 60 °C in the time-average peak fuel temperatures. Topics that will merit significant attention as the NGNP/AGR Fuel Program progresses include the evaluation of how such relatively large irradiation temperature uncertainties are (a) quantified, (b) affected by increasing thermocouple failures, and (c) conservatively treated in the contexts of fuel performance qualification and data development for use in developing and validating the analysis models for fuel radionuclide transport.

Item No.	Topic	Draft Assess Sec/Pg	February 2012 Draft Assessment (ML120240669)	July 2014 Final Assessment (ML14174A845)
17	Fuel	3.7.1/ 26	Applicability of Delayed Fuel Heating Tests: To assess the effects of delayed testing on fuel particle performance, a quantitative comparison of the respective inventories of all elements produced by fission, activation, and decay would first be needed to determine any substantial elemental inventory differences. This would then be used to assess how the respective differences in elemental inventories could potentially affect fuel particle performance and how fuel performance could be affected by other changes in fuel composition (e.g., species migration, chemical reactions, phase changes) that might be expected to occur during extended periods of post-irradiation cooling and decay. Assessment of the applicability of delayed fuel heating tests to fuel performance in HTGR accident conditions is a follow-up item.	The staff's assessment of the applicability of delayed fuel heatup testing proceeded during the follow-on assessment phase with DOE/INL's submittal of technical evaluation study TEV-1543. The NRC staff believes that the information provided in TEV-1543 adequately demonstrates the applicability of delayed fuel heatup testing to the evaluation of fuel performance in HTGR accidents involving fuel heatup either during or after at-power irradiation.

Item No.	Topic	Draft Assess Sec/Pg	February 2012 Draft Assessment (ML120240669)	July 2014 Final Assessment (ML14174A845)
18	Fuel	3.8/ 27	**Irradiation and Accident Proof Testing of Production Fuel:** Fuel produced by the NGNP Fuel Fabrication Facility (FFF) is likely to involve significant differences in the fuel production equipment, processes, and characterization methods. Accordingly, the working group believes that both irradiation proof testing and post-irradiation heating tests of fuel produced in the FFF should be conducted to demonstrate the acceptable performance of the FFF fuel and to qualify the FFF fuel for the NGNP reactor. It is anticipated that the FFF fuel irradiation qualification testing can be conducted on a schedule that would not adversely impact the NGNP prototype startup schedule.	It is expected that the fuel for the NGNP core will be fabricated in a large fuel fabrication facility with a number of production lines for fabricating fuel kernels, production lines with coaters for coating the fuel kernels, production lines for over-coating particles and production lines for making fuel compacts. Each production line is expected to produce fuel product in lots and batches. The variability (e.g., mean and standard deviation) of attributes of the finished fuel will depend on the variability across the lines and the way the lots and batches are mixed to feed into the next step in the fuel fabrication process. On the other hand, the fuel for fuel qualification (i.e., AGR-5/6) will likely be fabricated from a single line involving a single piece of fabrication equipment for each step in the fabrication process (i.e., kernel, coating, over coating and compacting). The attribute variability for fuel made on fabrication facility lines may differ from that for fuel made on a single production-scale line. In follow-on discussions concerning this issue, DOE/INL clarified its intent to avoid the need for proof testing by using mixed batches of fuel made on the single production-scale line for AGR-5/6 to simulate the variability of fuel made on the fuel fabrication facility lines for the NGNP prototype (ML12132A467). The technical basis for this variability simulation approach was not described in detail but necessarily relies in part on future activities of the NGNP/AGR Fuel Program and should therefore be evaluated and confirmed by the NRC staff when such activities have been completed.
26	Fuel	3.10.4/ 35	**Models and Data for Fuel Particle Performance During Reactivity Accidents:** Confirmation of the lack of a need for fuel particle failure data for NGNP reactivity insertion accidents is a follow-up item	The determination of fuel energy deposition and maximum fuel temperature for the most limiting NGNP reactivity insertion accidents depends on NGNP design and analysis details that have not been established. Until this information is developed and reviewed, the staff will not be able to assess whether needs exist for fuel testing specific to NGNP reactivity excursions.

Item No.	Topic	Draft Assess Sec/Pg	February 2012 Draft Assessment (ML120240669)	July 2014 Final Assessment (ML14174A845)
27	Fuel	3.10.5/ 36	**Models and Data for Accidents with Attack by Oxidants:** Chemical kinetics of graphite oxidants can be catalysed. Among the better catalysts are alkali metals and alkaline earths - that is, cesium and strontium that may have escaped the fuel particles and produced a "halo" around the fuel particles. One can conceive of preferential reactions at these catalyst sites that create pathways for rapid mass transport of oxidant to the fuel particles. It is not evident that catalysis of graphite oxidation has been considered in the analysis of either air or water intrusion accidents. The working group notes with interest the Project's statement on planned safety tests (radionuclide release at elevated temperatures) on compacts irradiated in graphite sleeves or on irradiated spherical fuel elements at various partial pressures of oxygen over a range of temperatures. The working group will remain interested in the status of these tests, and will welcome any information the Project can provide on water ingress testing. Evaluation of attack by oxidants is thus considered an item for follow up.	Chemical kinetics of graphite oxidants can be catalysed. Among the better catalysts are alkali metals and alkaline earths (i.e., cesium and strontium that may have escaped the fuel particles and produced a "halo" around the fuel particles). Preferential reactions could possibly occur at these catalyst sites that create pathways for the rapid mass transport of oxidants to the fuel particles. Consideration of the catalysis of graphite oxidation in the analysis of either air or water intrusion accidents is not evident. An analysis of an oxidant attack on matrix material should consider mass transport and include parameters for porosity, tortuosity, Knudsen permeability, and Poisseiulle permeability. The NRC staff notes DOE/INL's statement on planned safety tests (radionuclide release at elevated temperatures) on compacts irradiated in graphite sleeves or on irradiated spherical fuel elements at various partial pressures of oxygen over a range of temperatures. This statement was in response to an RAI on air ingress test plans for pebble and prismatic fuels. During the follow-on assessment phase, DOE/INL submitted a research plan that contained more detailed information on the experiments that it intends to perform for moisture and air ingress. The staff finds that the submitted experiment plan presents a reasonable approach for developing the data needed to model how air and moisture ingress can affect NGNP TRISO fuel performance and fission product transport. Ensuring that the experiments adequately envelop all LBEs that involve air or moisture ingress in the final NGNP design will be important.

Item No.	Topic	Draft Assess Sec/Pg	February 2012 Draft Assessment (ML120240669)	July 2014 Final Assessment (ML14174A845)
28	Fuel	3.10.5/ 37	**Effects of Air on Particle Coating Layers:** The planned integral safety tests of irradiated NGNP fuel at various partial pressures of oxygen over a range of accident temperatures are both appropriate and necessary to provide the need particle failure rate data for modeling particle failure during air ingress events. The experimental study of SiO2 formation versus SiO formation as a function of temperature and oxygen partial pressure is important in providing a qualitative and quantitative understanding and confirmation of the particle degradation phenomena for the integral test results. This is an item for follow up.	Item unchanged from the Draft Assessment Report.
44	Gen	3.13.2/ 46	**Design Features, Testing, and Surveillance Programs Specific to the NGNP Prototype:** The Project should specifically address how design features, testing, and surveillance programs specific to the NGNP prototype will be used to verify and supplement the developmental technical bases now being established for NGNP fuel qualification and mechanistic source terms. Such prototype-specific programs would entail the conduct of pre-operational, startup, and operational tests, operational monitoring and surveillance, and periodic confirmatory measurements and inspections. This is a follow-up item.	Item unchanged from the Draft Assessment Report.
45	Fuel	3.13.3/ 48	**Challenges and Needs for Verifying Normal Fuel Operating Conditions in HTGR Cores:** Inherent technical challenges make normal operating conditions in HTGR cores both difficult to measure and difficult to reliably predict. The Project should develop approaches and plans for performing in-core measurements in the NGNP prototype to verify normal core operating conditions and demonstrate the adequate detection of operating condition anomalies.	Item unchanged from the Draft Assessment Report.

46	Gen	3.13.4/ 50	**Prototype Testing and Surveillance:** The working group's view is that, going forward, a clearer understanding should be established regarding the full gamut of testing, monitoring, and surveillance programs and associated instrumentation systems envisioned for the NGNP prototype. Included would be a shared understanding of how such programs could be used to facilitate effective resolution of technical issues both generally and in the context of prototype licensing provisions. This would call for, among other things, periodic PIE and accident heatup testing on fuel discharged from the NGNP prototype, information on any advanced in-core detectors to be developed and deployed, and, more generally, information on how measurement data will be calibrated and used to (a) address technical specifications and (b) verify and supplement the developmental technical bases for NGNP fuel qualification and mechanistic source terms. This overall topic is considered a significant follow-up item.	Going forward, the staff believes that the NGNP applicant should establish a clearer understanding of the full spectrum of testing, inspection, monitoring, and surveillance programs and associated instrumentation systems envisioned for the NGNP prototype. In addition, the NGNP applicant should establish a shared understanding of how such programs could be used to facilitate effective resolution of technical issues both generally and in the context of prototype licensing provisions in accordance with 10 CFR 50.43(e)(2). Establishing such an understanding would require information on the development and deployment of any advanced in-core detectors and an explanation of how the NGNP applicant will calibrate and use measurement data to address technical specifications and to verify and supplement the developmental technical bases for NGNP fuel qualification and mechanistic source terms. For NGNP prototype licensing, the NRC would use conservatively evaluated pre-prototype-test fuel and core performance uncertainties as a basis for determining any additional requirements on design or operating parameters (e.g., staged trip set points, staged limits on core thermal power or core outlet temperatures) during the prototype testing period. To be useful as such, the NGNP prototype reactor module should be full-scale and functionally identical to the anticipated standard reactor module design. The NGNP prototype module may nevertheless need special design and operational provisions to accommodate the placement and removal of temporary probes and sensors in the core and primary system during the testing period. The NRC staff would expect the scope of NGNP prototype testing and surveillance to include at least the following: • Post-irradiation examination and accident heatup testing on used fuel discharged from the prototype. • Mapping of in-core and core-outlet temperatures during normal operation.

Item No.	Topic	Draft Assess Sec/Pg	February 2012 Draft Assessment (ML120240669)	July 2014 Final Assessment (ML14174A845)
				• Tests to establish or verify detection thresholds for plausible core operating anomalies (e.g., core hot spots caused by local obstructions of helium flow). Additional or optional prototype tests to confirm design and licensing analyses or reduce associated uncertainties may also include, among others: • Mapping of core and system temperatures under controlled or simulated conditions of loss of forced cooling or loss of coolant pressure. • Tests to further refine or validate selected fission product transport models.

APPENDIX B. STATISTICAL QC METHODS

The purpose of this appendix is to provide a brief summary of the principles of statistical testing that are applied in the quality characterization of coated fuel particles and spherical fuel elements.

Given that the quantities of fuel particles in the reactor core number in the billions, acceptance testing of the fuel particles and fuel compacts to determine conformance to specification requirements is necessarily performed on a statistical basis (i.e., statistical quality control). Acceptance testing for fuel spheres is similarly subject to statistical testing, although the number of items on which to perform acceptance testing is several orders of magnitude lower. The statistical methods are summarized below, and the risks of false acceptance and false rejection that are inherent in statistical sampling are discussed.

The basic approach is to collect a representative sample, apply an acceptance test to the sample, and accept or reject the product based on this test. Even though a sample satisfies the acceptance test, it is not certain that the population meets the acceptance criteria. A chance of a wrong decision always exists when the decision is based on a random sample, but this can be quantified and made small.

Two types of wrong decisions can be made. The first is made when a product that does not meet the specifications is accepted (false acceptance). A test corresponds to c% confidence when any unacceptable product has at most a (100 - c)% chance of being accepted. For example, if the acceptance test has a 95% confidence level, there is no more than a 5% chance of accepting a product that should be rejected.

The second type of wrong decision is made when an acceptable product is rejected (false rejection). The risk associated with making this decision must also be minimized. For a fixed sample size, the two kinds of wrong decisions are inversely related. As the sample size increases, both risks decrease.

The product specifications establish acceptance criteria for the properties of concern as well as the confidence levels with which the population must meet the criteria. A property may be stated in terms of kernels, coated particles, or compacts. Each property is one of two types: attribute or variable.

An attribute property is discrete in the sense that, for example, the particle is either defective or not in terms of that property. For example, “missing buffer” is an attribute property—either the buffer is present, or it is missing. The acceptance criterion for an attribute property is stated in terms of the allowable fraction of defective particles. To test whether the population satisfies a criterion for an attribute property, a sample is drawn, and each item is found to be either acceptable or defective in terms of the criterion. If the number of defective items is small enough, the population is accepted. The numbers defining the test are adjusted so that the probability of a false acceptance is, at most, 5%.

A variable property is a property defined by a continuous distribution, such as the normal distribution. The acceptance criterion for a variable property is stated in terms of the population mean and/or the population dispersion. For the population mean, the criterion is that the mean must lie within a specified interval. In some instances, this interval is one-sided. The endpoints of the interval are the acceptance limits for the mean. For population dispersion, the criterion is that no more than a specified fraction of the population can exceed and/or be less than predetermined values. These values are called the critical limits for the dispersion.

A confidence interval is used to test whether the population meets a criterion for the mean. There are two ways to test whether the population meets a criterion for dispersion. The first is to assume that the population is normally distributed and to base the test on the sample mean, the sample standard deviation, and tabulated tolerance factors for the normal distribution. The second way, if the normality assumption is not justified, is to treat the property as an attribute property, in which case the particle is deemed defective if it exceeds the critical limit and is acceptable otherwise. The test is based on the number of defective items found in a sample. The consequence of not assuming normality is that a considerably larger sample size is required.

An acceptance test is simply a decision rule for determining acceptance or rejection of a population based on a sample. When testing for a variable property based on the population mean, and given a two-sided acceptance criterion, the quantities A and B are calculated:

$$A = \bar{X} - ts/\sqrt{n} \text{ and } B = \bar{X} + ts/\sqrt{n} \text{ where}$$

$\bar{X}$ = sample mean,

s = sample standard deviation,

t = Student's t value based on sample size and confidence level,

and n = sample size.

If $A < L_m$, the lower acceptance limit for the mean, or $B > Um$, the upper acceptance limit for the mean, the population is rejected. Otherwise, the population is accepted based on the sample. The 95% confidence level is achieved by use of the 95th percentile of Student's t distribution, with n-1 degrees of freedom. For a one-sided interval, the method is the same, except that only A or B applies. The 95th percentile of Student's t ensures that an unacceptable population is rejected with probability 0.95.

The acceptance test of a variable property based on the population dispersion, using the assumption of normality, is as follows. Given a two-sided acceptance criterion, the quantities C and D are calculated:

$$C = \bar{X} - ks \text{ and } D = \bar{X} + ks$$

where

$\bar{X}$ = sample mean,

s = sample standard deviation, and

k = tolerance factor for normal distribution based on sample size, confidence level, and allowable fraction outside of critical limit

If $C < L_p$, the lower critical limit for the population, or $D > U_p$, the upper critical limit for the population, the population is rejected. Otherwise, the population is accepted. For one-sided acceptance criteria, the method is the same except that only C or D applies. The confidence level is achieved by the appropriate use of the tolerance factor, k. One-sided tolerance factors are used for both one- and two-sided acceptance criteria. For testing attribute properties, a double sampling plan is generally employed. Under

such a plan, the first sample is drawn and inspected. If the first sample is sufficiently good (in terms of the number of defective particles), the product is accepted without further testing. If it is sufficiently poor, the product is rejected. If the sample falls into neither category, a second sample is taken and inspected before the decision is made. For acceptance testing of product populations for attribute properties, the sample sizes are based on binomial probabilities to ensure a <5% false acceptance rate.

A variable property may be treated as an attribute property and subjected to the type of test described above if the normality assumption is not justified. One such property is the sphericity (aspect ratio). It is ideally 1.0; it may be larger, but it cannot be smaller. Therefore, it has an asymmetrical distribution rather than a normal distribution. To treat sphericity as an attribute property, a particle is counted as defective if the particle's aspect ratio is greater than the allowed limit and counted as acceptable if its aspect ratio is less than the allowed limit.

Table B-1 provides an example of the relationship between the sample size and acceptance criterion for an attribute test based on the binomial distribution. This example is for an attribute having an acceptance limit of 5.0 × 10-5.

Table B-1. Acceptance Number vs. Sample Size for Attribute Property Acceptance Test

Sample Size	Max. Number of Defects for Acceptance at 95% Confidence Level	Indicated Defect Level
59914	0	0
94876	1	1.1×10^{-5}
183068	4	2.2×10^{-5}
314101	9	2.9×10^{-5}
996164	38	3.8×10^{-5}

As noted above, two kinds of acceptance test errors may occur—false acceptance, which is acceptance of product that does not actually satisfy the required criteria, and false rejection, which is rejection of product that actually satisfies the criteria. These errors can occur because each decision to accept or reject is based on a random sample, not on examination of the entire population. The probability of each kind of error can be found by calculating the probability of rejecting the population, assuming that the population is specified. Ideally, the probability of rejection should be large for an unacceptable population and small for an acceptable population. This is accomplished by careful selection of the sample size.

As an example, consider the acceptance criterion for the mean kernel diameter of a composite. In this particular example, the acceptance criterion requires that the population mean be less than the upper acceptance limit of 360 µm and greater than the lower acceptance limit of 340 µm. Figure B-1 shows the probability of rejection (power curve) of the composite for the kernel diameter criterion as a function of sample mean value and sample size, assuming a standard deviation of 10 µm. The specifications on the mean are shown as vertical lines at 340 and 360 µm. The figure shows that for a small sample size (n = 10), the probability of rejection when the true mean is within the specifications is unacceptably high. For example, if the true mean is 355 µm, the probability of rejecting the composite based on a sample size of 10 is more than 0.50. If the sample size is increased to 50, the probability of rejecting a composite with a

true mean of 355 µm is less than 0.05. This suggests that a sample size of at least 50 kernels should be collected for this property.

Table B-1 and Figure B-1 illustrate a very important aspect of statistical QC. Specifically, as the true value of a property in a population that is within specification with respect to that specification approaches the specification limit, the minimum sample size that will be needed to accept the population at the 95% confidence level (and to avoid false rejection of the population) becomes quite large. Indeed, for testing of an attribute such as the SiC defect fraction, essentially 100% inspection would be required if the actual value of the SiC defect fraction is just below the specification limit.

The most important point to be taken from this discussion is that the economics of fuel manufacturing dictate that the fuel manufacturer must strive to achieve a quality level that is significantly better than specification requirements to avoid excessive rejection of good product with reasonable sample sizes. What this means is that the necessary use of statistical QC for acceptance testing of HTGR fuel effectively guarantees that the average fuel quality delivered by the fuel manufacturer to the reactor will significantly exceed that required by the fuel-product specifications.

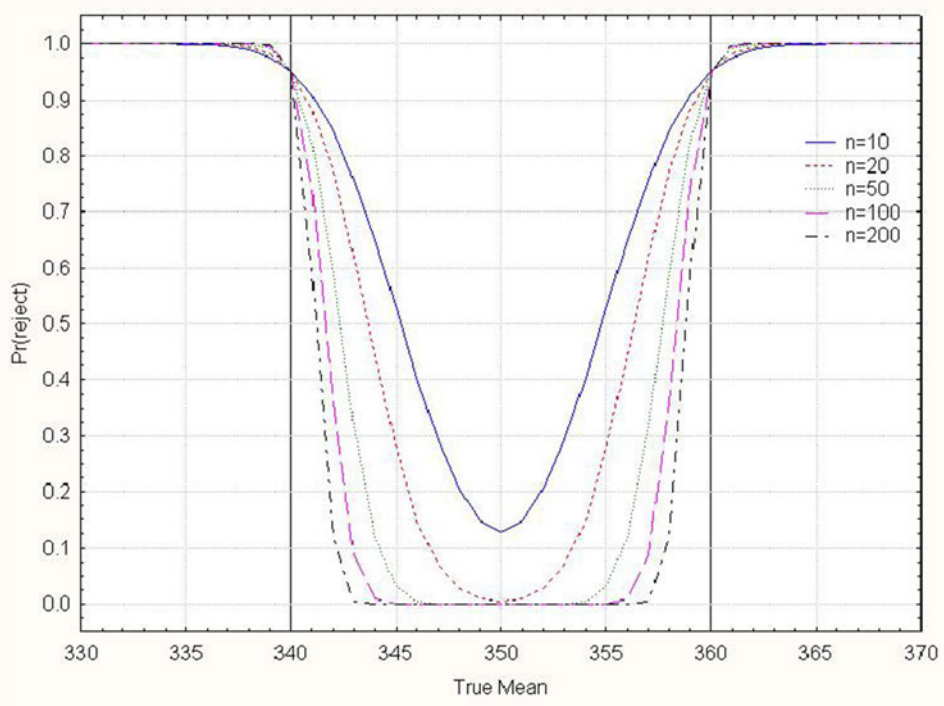

Figure B-1. Probability of Rejecting Composite for Mean Kernel Diameter

APPENDIX C. FUEL SPHERE QUALITY TESTING

The purpose of this appendix is to provide a brief summary of the quality testing methods used to confirm the important characteristics of the fabricated spherical fuel elements.

The following fuel-sphere characteristics are monitored. Some characteristics are determined on fuel-free spheres made from the same lot of material as the fuel spheres.

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