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Plan

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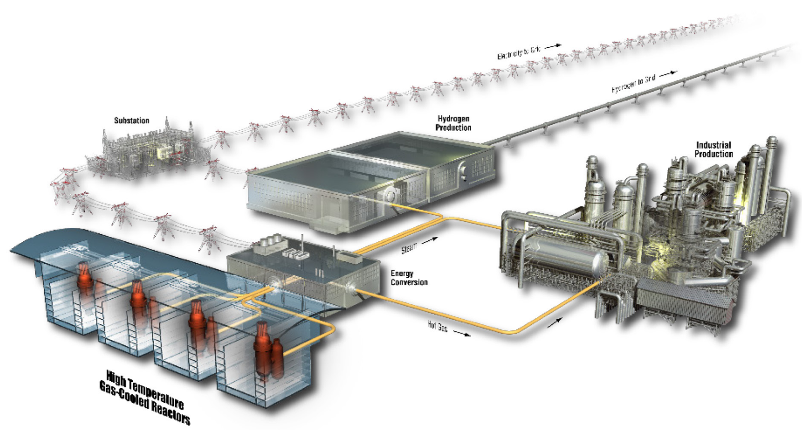
Technical Program Plan for INL Advanced Reactor Technologies Advanced Gas Reactor Fuel Development and Qualification Program



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REVISION LOG

Rev.	Date	Affected Pages	Revision Description
0	09/30/2010	All	New issue, refer to those below for earlier versions: INL/EXT-05-00465, Revision 2, July 2008 INL/EXT-05-00465, Revision 1, August 2005
1	12/03/2012	All	Updated to incorporate changes to project planning and complete biennial review of document.
2	12/18/2012	4, 8, 9, 42, 43, 44, 51, 52, 53, 56, 57, and 64	Revised to incorporate INL Regulatory Affairs and NRC agreements and commitments.
3	05/05/2014	All	Revised/updated per routine review.
4	05/07/2015	All	Revised/updated per routine review.
5	05/09/2016	All	Revised/updated per routine programmatic review.
6	06/28/2017	All	Revised/updated per routine review.
7	06/29/2018	All	Revised/updated per routine review.
8	06/28/2019	All	Revised/updated per routine review.
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SUMMARY

High-temperature gas-cooled reactors (HTGRs) are graphite-moderated nuclear reactors cooled with helium. Their high outlet temperatures and thermal energy conversion efficiency enable cost-effective integration with non-electricity-generating applications. These applications include process heat and hydrogen production for petrochemical and other industrial processes that require operating temperatures between 300 and 900°C. HTGRs will supplement the use of premium fossil fuels, such as oil and natural gas, to improve overall energy security in the United States (U.S.) by reducing dependence on foreign fuels. They will also reduce carbon dioxide (CO_2)/greenhouse gas emissions. The HTGR design uses helium as a coolant, graphite as a neutron moderator, and ceramic particle fuel. Helium is chemically inert and neutronically transparent. The graphite core slows down the neutrons, retains its strength at high temperatures, provides structural stability, and acts as a substantial heat sink during transient conditions. The ceramic particle fuel is extremely robust and retains the radioactive byproducts of the fission reaction within the coated particle under normal and off-normal conditions.

The U.S. Department of Energy (DOE) Office of Nuclear Energy (NE) and Idaho National Laboratory (INL) Advanced Reactor Technologies (ART) Advanced Gas Reactor (AGR) Fuel Development and Qualification Program (referred to hereafter as the AGR Fuel Program) are pursuing qualification of tristructural-isotropic (TRISO) particle fuel for use in HTGRs. The AGR Fuel Program was established to achieve the following overall goals:

- Provide a fuel qualification dataset to support the licensing and operation of an HTGR. HTGR fuel performance demonstration and qualification comprise the longest-duration research and development (R&D) tasks required for design and licensing. The fuel form is to be demonstrated and qualified for service conditions that include normal operation and potential accident scenarios.
- Support deployment of HTGRs for hydrogen, process heat, and energy production in the U.S. by reducing market entry risks posed by technical uncertainties associated with fuel production and qualification.
- Extend the value of DOE NE resources by using international collaboration mechanisms where practical.
- Support establishment of a domestic TRISO particle fuel manufacturing capability for fabricating demonstration and qualification experiment fuel.

TRISO particle fuel development and qualification activities support prismatic and pebble-bed HTGR fuel designs. The AGR Fuel Program has focused to date on manufacturing and testing the fuel design for HTGR concepts using the most recent gas-turbine modular-helium reactor fuel product specification[1] as a starting point. Irradiation, safety testing, and post-irradiation examination (PIE) plans will support fuel development and qualification in an integrated manner. Preliminary operating conditions and performance requirements for the fuel, as well as preliminary fuel product specifications to guide the AGR Fuel Program's fuel fabrication process development activities, are based on previously completed HTGR design and technology development activities, operating conditions, and performance requirements.

At the onset of the AGR Fuel Program in 2002 (then known as the Advanced Gas Reactor Fuel Development and Qualification Program), facilities and personnel experienced in relevant activities existed in the U.S., primarily at INL and Oak Ridge National Laboratory (ORNL). INL and ORNL personnel with experience and knowledge of TRISO particle fuel, facility status, and capabilities were involved in developing the initial Technical Program Plan for the Advanced Gas Reactor Fuel Development and Qualification Program.[2] In addition, General Atomics provided input regarding prismatic HTGR fuel

performance requirements and perspectives from its experience in fuel development, fuel fabrication, and fuel-related analytical capabilities needed to support licensing interactions. BWX Technologies, Inc. (BWXT-NOG) also provided input based on its experience and capabilities for fuel kernel production and fuel particle coating. Many of the individuals who helped develop this plan were directly involved in producing and testing previous U.S. fuel for the modular high-temperature gas-cooled reactor and the new production reactor. They conducted extensive investigations and reviews in the early 1990s following the unexpectedly high fuel failure levels observed in those tests. This plan builds directly on the large body of coated particle fuel experience and is generally consistent with the recommendations arising from those experiences.

Based on the recommendation of the Nuclear Energy Advisory Committee to Congress in 2011, design-specific efforts on the Next Generation Nuclear Plant (NGNP) project were halted at the end of the conceptual design phase in 2011, in part because a viable public-private partnership for a demonstration plant and follow-on commercialization had not yet been established. With no HTGR reactor deployment anticipated in the near term at the time, it was decided that the R&D program focus would be to qualify a fuel and establish a commercial fuel vendor in the U.S. There has not been, nor will there be, an effort to verify or validate any potential reactor vendor codes as a part of the HTGR R&D performed under the AGR Fuel Program. With the exception of the AGR-3/4 irradiation and post-irradiation examination (PIE), the effort to quantify fission product transport within reactor core materials and provide a technical basis for the source term has similarly been halted after initial hydrogen and tritium permeation testing in various stainless steel alloys.

The AGR Fuel Program involves five major program elements:

1. Fuel Fabrication. This program element, to fabricate successful TRISO particle fuel (which must meet the fuel quality and performance requirements for licensing an HTGR), requires developing a coating process that replicates, to the greatest extent possible, the HTGR particle design and properties of the coatings on German fuel particles that have previously exhibited superior irradiation and accident performance. Coating process development has been accomplished in two phases: initially in a 2-in.-diameter, laboratory-scale coater (AGR-1) followed by scale-up to a 6-in., prototypic, production-scale coater (AGR-2). The Fuel Fabrication program element has included establishing the fuel fabrication infrastructure; developing the process for the low-enriched uranium carbide/oxide kernels, TRISO particles, and compacts; developing coating process models; developing quality control methods; performing fuel process scale-up analyses; and developing process documentation for technology transfer to private industry. The fuel-fabrication effort produces TRISO particle fuel within cylindrical fuel compacts that meets fuel product specifications and provides fuel and material samples for characterization, irradiation, safety testing, and PIE as necessary to meet the overall AGR Fuel Program goals.

2. Fuel and Material Irradiation. This program element provides data on fuel performance during irradiation to support fuel process development, qualify a fuel design and fabrication process for normal operating conditions, and support development and validation of fuel performance and fission product transport models and codes. This program element also provides irradiated fuel and materials necessary for PIE and Safety Testing. Seven irradiation tests, designated as AGR-1 through AGR-7, have been defined to provide data and sample materials within the AGR Fuel Program.

3. Fuel PIE and Safety Testing. This program element provides the facilities and processes to measure the performance of TRISO particle fuel under normal operating and potential accident conditions. Moisture and air ingress testing in quantities expected to exist within the typical helium and neon gas supplies used during irradiation (testing performed during AGR-3/4 irradiation) and safety testing

(planned to be performed during AGR-5/6/7 PIE) will be performed to determine their effects on TRISO particle fuel. This work supports the fuel manufacturing effort by providing feedback on the performance of kernels, coatings, and compacts during irradiation and under potential accident conditions. PIE and Safety Testing provide a broad range of data on fuel performance and fission product transport within TRISO-coated fuel particles, compacts, and graphite materials representative of fuel element blocks. These data, in combination with the in-reactor measurements (irradiation conditions and fission gas release-rate-to-birth-rate ratios), are necessary to demonstrate compliance with fuel performance requirements and support developing and validating computer codes.

4. Fuel Performance Modeling. This program element addresses the structural, thermal, and chemical processes that can lead to TRISO-coated particle failures. It considers the effects of fission product chemical interactions with the coatings, which can lead to degradation of the coated particle properties. Fission product release from the fuel particles and transport in the fuel compact matrix and fuel element graphite during irradiation are also modeled. Computer codes and models will be further developed and validated as necessary to support fuel fabrication process development.

5. Fission Product Transport and Source Term. This program element addresses the transport within reactor core materials of fission products produced in the TRISO particle fuel. It is intended to provide a technical basis for source terms for HTGRs under normal irradiation and potential accident conditions. However, most of this work scope has not been performed because of funding shortfalls and higher-priority work scopes. Some initial fission product transport studies were performed on hydrogen and tritium permeation through high-nickel superalloys with results that were included in published reports. An evaluation of data from irradiation and safety testing of “designed-to-fail” fuel particles will be performed as part of the AGR-3/4 PIE. The purpose of the evaluation is to characterize fission product release and transport from TRISO particle fuel into fuel compact matrix and fuel element graphite under normal and off-normal HTGR conditions.

This plan aims to develop an understanding of the relationships among the fuel fabrication process, fuel product properties, and irradiation and safety test performance. Precise process control, as well as advanced characterization and data-acquisition methods, conducted within a structured quality assurance framework, are important elements in achieving this objective. Qualified fuel performance data produced under fuel irradiation conditions and in-pile gaseous fission product release, as well as a wide range of data produced during PIE and Safety Testing, are important elements. Fuel performance modeling is also included. The fuel performance models are considered essential for several reasons, including (1) guidance for a future plant designer/applicant in establishing the reactor core design and operating limits and (2) demonstrating to the licensing authority, the U.S. Nuclear Regulatory Commission (NRC), that the applicant has a thorough understanding of the in-service behavior of the fuel system.

The five program elements and the activities associated with each are discussed in Section 3 of this technical program plan. Early AGR Fuel Program activities were centered on the fuel fabrication element because the production of fuel and materials for irradiation, safety testing, and PIE were the early critical-path activities. Now the critical-path activity is the performance of PIE and Safety Testing for each of the remaining experiments.

Key accomplishments of the AGR Fuel Program to date are listed below:

- Developed low-enriched uranium (LEU) carbide/oxide TRISO fuel fabrication and modern quality control capabilities, first at laboratory scale and then at pilot scale at a domestic vendor facility. Outcomes include:
  - Improved kernel forming, carbothermic reduction, and sintering chemistries

  - Thirty-fold increase in TRISO particle capacity in the coating furnace
  - Improved methods of producing and applying resinated graphite powder overcoats to TRISO particles that eliminate multiple process steps, eliminate Resource Conservation and Recovery Act (RCRA) mixed-hazardous waste generation, and reduce production time by one order of magnitude
  - Demonstration of a multi-cavity, fully automated compacting press
  - Demonstration of a combined-cycle thermal treatment process for finishing compacts

- Completed fabrication and delivery of the AGR-5/6/7 fuel compacts.
- Developed test train designs for multi-capsule individual and multi-experiment tests.
- Completed three irradiation campaigns, which included the first four AGR experiments: AGR-1 for 620 effective full-power days (EFPDs) with no particle failures; AGR-2 for 559 EFPDs with no apparent particle failures; and AGR-3/4 for 369 EFPDs containing designed-to-fail TRISO particles that failed during irradiation.
- Completed assembly and initiated irradiation of the AGR-5/6/7 experiment in the Advanced Test Reactor (ATR) in February 2018; achieved 175 EFPDs as of June 2019.
- Completed PIE and safety testing of the AGR-1 irradiated fuel and components, including inert-atmosphere safety tests on 19 compacts, most of which were isothermal tests at temperatures of 1600, 1700, or 1800°C for approximately 300 hours each. In one of the tests, three compacts were simultaneously tested using a varying temperature profile (e.g., a minimum temperature 830°C, a maximum temperature 1690°C) simulating a temperature transient during a core-conduction cool-down event.
- Completed disassembly of the AGR-2 and AGR-3/4 test trains and capsules, and completed metrology and gamma scanning of the capsule components and fuel compacts at INL.
- Initiated destructive PIE of thirteen as-irradiated AGR-2 compacts, including deconsolidation-leach-burn-leach analysis and follow-on particle exams on nine compacts (eight uranium carbide/oxide [UCO] and one uranium dioxide [UO_2]) and cross section analysis of four compacts (three UCO and one UO_2).
- Completed ten 1600°C safety tests at ORNL of AGR-2 compacts (two UO_2 and eight UCO). Completed one 1700°C safety test of an AGR-2 UO_2 compact. Completed five 1800°C safety tests of AGR-2 compacts (two UO_2 and three UCO).
- Established capability to re-irradiate loose particles in the INL Neutron Radiography (NRAD) reactor; currently developing the capability to re-irradiate whole AGR-3/4 compacts. Together, with subsequent safety testing of the re-irradiated fuel, this enables study of I-131 release from AGR fuels, a critical part of the fuel safety analysis.
- Developed equipment and methods for destructive PIE at INL of AGR-3/4 graphite rings containing fission products and completed sampling of inner and outer rings from four capsules.
- Installed necessary equipment in radiation hot cell and performed radial deconsolidation of the first five irradiated AGR-3/4 compacts at INL.

- Completed conceptual design review, initiated equipment procurement/construction, and completed final design review of furnace and associated systems for safety testing irradiated fuels under air/moisture ingress conditions at INL.
- Developed the fuel performance modeling code, Particle Fuel Model (PARFUME), which has been used for irradiation pre-test and safety test predictions and refined based on information from AGR-1 PIE.
- Completed initial hydrogen and tritium fission product transport permeation of stainless steel alloy studies.
- Established the Nuclear Data Management and Analysis System database for collection and management of data obtained during fuel fabrication, irradiation, PIE, and safety testing.
- Collected, analyzed, and qualified millions of data points generated during fuel fabrication, irradiation, and PIE for future support of NRC licensing activities of TRISO particle fuel.

In addition, in 2014, the NRC staff completed its assessment of two NGNP licensing white papers titled "NGNP Fuel Qualification White Paper"[3] and "Mechanistic Source Terms White Paper."[4] These papers described the AGR Fuel Program and its approach to determining mechanistic source terms, which relied extensively on data obtained from the AGR Fuel Program. The results of the NRC's assessment were documented and transmitted to DOE via a letter with two enclosures.[5] The enclosures provided feedback on key licensing issues that are closely tied to the AGR Fuel Program, the approach to fuel development and qualification, and to mechanistic source terms. These significant NRC findings indicate that the AGR Fuel Program is on track to meet its goal of providing a fuel qualification dataset in support of the licensing and operation of an HTGR.

More recently, AGR program staff collaborated with EPRI to prepare a topical report on UCO TRISO fuel performance based on results from the AGR-1 and AGR-2 fuel fabrication, irradiation, PIE, and safety testing activities[38]. The report describes the historic basis for the current TRISO particle design and provides AGR data demonstrating the performance of UCO TRISO fuel particles under the range of conditions enveloped by the AGR-1 and AGR-2 irradiations and safety tests. The NRC is currently evaluating the report, and a Safety Evaluation is expected by the end of FY20. This will accelerate licensing of advanced high-temperature reactors by providing NRC concurrence that TRISO particles fabricated to the AGR specifications will have similar performance as demonstrated by the AGR-1 and AGR-2 fuel under similar service conditions.

CONTENTS

FIGURES

TABLES

ACRONYMS

AGC	Advanced Graphite Creep
AGR	Advanced Gas Reactor
ART	Advanced Reactor Technologies
ATR	Advanced Test Reactor
AVR	Arbeitsgemeinschaft Versuchsreaktor
BWXT-NOG	BWX Technologies Nuclear Operations Group
CCCTF	Core Conduction Cooldown Test Facility
CO	carbon monoxide
CO_2	carbon dioxide
Cs	cesium
DOE	U.S. Department of Energy
DOE-NE	U.S. Department of Energy Office of Nuclear Energy
DTF	designed-to-fail
EDF	Engineering Design File
EDS	energy-dispersive x-ray spectroscopy
EFPD	effective full power day
FACS	fuel accident condition simulator
FCF	Fuel Conditioning Facility
FIMA	fissions per initial heavy metal atom
FY	fiscal year
GA	General Atomics
GIF	Generation IV International Forum
HFEF	Hot Fuel Examination Facility
HFIR	High Flux Isotope Reactor
HRB	High Flux Isotope Reactor Removable Beryllium
HTGR	high-temperature gas-cooled reactor
HTR	high-temperature reactor
IAEA	International Atomic Energy Agency

INL	Idaho National Laboratory
IPyC	inner pyrolytic carbon
LBL	leach-burn-leach
LCB	life-cycle baseline
LEU	low-enriched uranium
MFC	Materials and Fuels Complex
MHTGR	modular high-temperature gas-cooled reactor
NE	Office of Nuclear Energy
NEFT	northeast flux trap
NGNP	Next Generation Nuclear Plant
NPR	new production reactor
NQA	Nuclear Quality Assurance
NRAD	Neutron Radiography Reactor
NRC	Nuclear Regulatory Commission
OD	outside diameter
OPyC	outer pyrolytic carbon
ORNL	Oak Ridge National Laboratory
PARFUME	Particle Fuel Model
PIE	post-irradiation examination
PyC	pyrolytic carbon
QA	quality assurance
QC	quality control
R&D	research and development
R/B	release-rate-to-birth-rate ratio
RCRA	Resource Conservation and Recovery Act
TRISO	tristructural-isotropic
UCO	uranium carbide/oxide
UO_2	uranium dioxide

VHTR	very high temperature reactor
WBS	work breakdown structure

1. INTRODUCTION

High-temperature gas-cooled reactors (HTGRs) are graphite-moderated nuclear reactors cooled with helium. Their high outlet temperatures and thermal energy conversion efficiency enable efficient, cost-effective integration with non-electricity-generation applications. These applications include process heat and/or hydrogen production for petrochemical and other industrial processes that require temperatures between 300 and 900°C. HTGRs will supplement the use of premium fossil fuels, such as oil and natural gas, improve overall energy security in the U.S. by reducing dependence on foreign fuels, and reduce carbon dioxide (CO_2)/greenhouse gas emissions. Key characteristics of the HTGR design include using helium as a coolant, graphite as a neutron moderator, and ceramic particle fuel. Helium is chemically inert and neutronically transparent. The graphite core slows down the neutrons, retains its strength at high temperatures, provides structural stability, and acts as a substantial heat sink during transient conditions. The ceramic particle fuel is extremely robust and retains the radioactive byproducts of the fission reaction under normal and off-normal conditions.

The U.S. Department of Energy (DOE) Office of Nuclear Energy (NE) has selected the HTGR as a transformative application of nuclear energy that will demonstrate emissions-free nuclear-derived electricity, process heat, and hydrogen production. The first-of-a-kind HTGR envisioned extends past applications of gas-cooled reactor technologies and will be driven by near-term commercial industry needs and current technology availability. The reference concept will be an HTGR with a design goal outlet gas temperature of 750 to 800°C. The reactor core may be either a prismatic graphite-block core or a pebble-bed core. The reactor fuel concept will use low-enriched uranium (LEU) to obtain high burnup in a "once-through" fuel management scheme.

In developing the original version of the technical program plan, priority was given to early activities in support of near-term execution. Issues associated with longer-term activities are being addressed in more detail as they arise, and their impact is being factored into overall planning. This additional detail has not affected the basic logic of the plan but does affect the details of its execution. Based on the coordinated planning activities discussed previously, the initial technical program plan[2] was issued by Oak Ridge National Laboratory (ORNL) in April 2003. Maintenance of the planning documentation was assigned to Idaho National Laboratory (INL) in 2004, consistent with its lead management role in the Advanced Reactor Technologies (ART) Advanced Gas Reactor (AGR) Fuel Development and Qualification Program (hereafter referred to as AGR Fuel Program). This plan has continued to be updated periodically to reflect additional knowledge and the results of ongoing and completed work. After the initial release, the next two revisions of the plan were issued as external documents under INL document control protocol and entitled INL/EXT-05-00465, *Technical Program Plan for the Next Generation Nuclear Plant/Advanced Gas Reactor Fuel Development and Qualification Program*, Revisions 1 and 2. The documentation protocol was changed within INL in 2010, which explains the current designation as a plan document (PLN-3636). Plan execution is adjusted according to progress, results, funding changes, and limitations in terms of milestones, completion dates, and work scope. Routine revisions to the plan are issued based on the actual funding received, accomplishments, and changes in technical directions as they evolve.

1.1 Program Scope and Background

In fiscal year (FY) 2002, the DOE Office of Nuclear Energy, Science, and Technology initiated development of the AGR Fuel Program for coated particle fuel. The resulting *Technical Program Plan for Advanced Gas Reactor Fuel Development and Qualification Program* and subsequent revisions defined fuel development activities to support licensing and operating an HTGR in the U.S. under the umbrella of the Next Generation Nuclear Plant (NGNP) project in accordance with the Energy Policy Act of 2005.

Based on the recommendation of the Nuclear Energy Advisory Committee to Congress,[6] design-specific efforts on the NGNP project were halted in 2011 at the end of the conceptual design phase, in part because a viable public-private partnership for a demonstration reactor and follow-on commercialization had not yet been established. Currently, no partnership has been formed, although several private companies have recently expressed interest in using the HTGR concept in an advanced reactor design. With no HTGR deployment anticipated in the near term, the research and development (R&D) program focus is to qualify a fuel form and support establishment of a commercial fuel vendor in the U.S. The HTGR R&D will not perform verification or validation of any potential reactor vendor codes.

This latest revision of the technical program plan describes the updated path forward for developing and qualifying tristructural-isotropic (TRISO)-coated particle fuel that incorporates the experience and knowledge gained from ongoing and completed work. HTGR designs provide inherent safety, which prevents core damage under nearly all design basis accidents and hypothetical severe accidents. The principle guiding this concept is to maintain core temperatures passively below fission product release thresholds under all potential accident scenarios. The required level of fuel performance and fission product retention reduces the radioactive source term at the reactor core boundary by many orders of magnitude and, relative to the core inventory, allows potential elimination of the need for evacuation and sheltering beyond a small exclusion area. This safety approach, however, mandates exceptional fabricated fuel quality and fuel performance under normal operating and potential accident conditions. Germany produced and demonstrated high-quality fuel for their pebble-bed reactors in the 1980s, but no U.S.-fabricated fuel had exhibited equivalent performance prior to the AGR Fuel Program. As in many reactor technology development programs, fuel development and qualification were identified as essential to ensure concept viability.

A complete set of fuel design specifications for an HTGR is not available to the AGR Fuel Program, but the maximum burnup envisioned in a prismatic HTGR is within the range of 150 to 200 GWd/metric tons of heavy metal or 16.4 to 21.8% fissions per initial heavy metal atom (FIMA). Maximum burnups for pebble-bed designs are typically considerably less than this. Although Germany has demonstrated excellent performance of uranium dioxide (UO_2) TRISO particle fuel up to about 10% FIMA and 1150°C, UO_2 fuel is known to have limitations because of carbon monoxide (CO) formation; kernel migration at the higher burnups; and power densities, temperatures, and temperature gradients that may be encountered in the prismatic HTGR design. With uranium carbide/oxide (UCO) fuel, the kernel composition is engineered to minimize CO formation and kernel migration, which are key threats to fuel integrity at higher burnups, temperatures, and temperature gradients. Furthermore, the performance of German silicon carbide (SiC)-based, TRISO-coated particle UCO fuel is up to 22% FIMA (as measured by the in-pile gas release in irradiation test FRJ2-P24[7]). The excellent performance of U.S.-made UCO fuel in AGR-1 and AGR-2 provides added confidence that high-quality SiC-based, TRISO-coated particle UCO fuel can be made and that its superior irradiation performance can be statistically demonstrated.

In addition to excellent fission product retention during normal operation at high burnups and high temperatures, HTGR fuel must exhibit satisfactory fission product retention under postulated accident conditions. Limited data on the accident performance of SiC-based TRISO-coated UO_2 fuel at high burnups indicate increased cesium (Cs) releases at burnups ≥14% FIMA,[8] so safety testing is an important element. The AGR Fuel Program chose to develop coated particle fuel using a low-enriched UCO kernel to qualify a fuel to meet fuel performance requirements under specified fuel service conditions. Thus, SiC-based TRISO-coated UCO was chosen as the baseline AGR fuel to be fabricated and tested. This fuel development path complemented particle fuel development with a UO_2 kernel that was being pursued by South Africa, China, and Europe at the time. Safety testing of irradiated AGR-1 and AGR-2 UCO TRISO compacts has demonstrated robust behavior for about 300 hours at 1600, 1700, and 1800°C, providing added confidence that SiC-based TRISO particle fuel can meet safety performance requirements.

The TRISO-coated UCO fuel specification[1] utilizing SiC as the primary fission product retention layer was developed in response to extensive evaluations[9,10] of the fuel failures experienced in irradiations in the New Production Reactor (NPR) and the Modular High-Temperature Gas-Cooled Reactor (MHTGR) Programs. This was the starting point for the fuel specification developed for the current program.[11] It is expected that this fuel will exhibit acceptable fuel performance at higher burnups (16 to 22% FIMA) at time-averaged fuel temperatures up to at least 1250°C for normal operation and 1600°C for potential accident conditions. The fuel is also expected to exhibit fast neutron fluences up to at least 5×10^{25} neutrons/m^2. This plan identifies R&D needed in the areas of fuel fabrication, fuel and materials irradiation, safety testing and post-irradiation examination (PIE), fuel performance modeling, and fission product transport and source term studies. Section 4 provides an updated integrated schedule and budget for the work required to develop, scale up to production capability, and transfer the TRISO particle fuel fabrication capability to an industrial fuel vendor within the U.S.

In the late 1980s, coated particle fuel performance was demonstrated at the desired level of quality and predictability in the Arbeitsgemeinschaft Versuchsreaktor (AVR) at Jülich, Germany, and several materials test reactors. The AGR Fuel Program has used a fuel design based on the most recent gas turbine modular helium reactor fuel product specification, combined with successful German-like coating and matrix material overcoating processes. The basic structure of the AGR Fuel Program is delineated in the major program elements below:

- Fuel Fabrication:
  - Develop capabilities for fuel fabrication at laboratory scale for establishing and refining the processing parameters.
  - Develop fuel fabrication capabilities at the prototypic production scale.
  - Develop a modern suite of characterization and quality control (QC) methods.
  - Transfer the fuel fabrication and QC technology to an industrial/commercial domestic fuel vendor.
  - Produce final reference fuel with a prototypic production-scale coater for fuel qualification testing.
- Fuels and Materials Irradiation:
  - Develop multi-capsule irradiation test train designs for individual and multi-experiment tests.
  - Develop fission gas monitoring systems and provide real-time measurement of fission gas released from each of the irradiation experiments.
  - Complete irradiation of the AGR-1 experiment for approximately 600 effective full power days (EFPDs), which is the initial shakedown test.
  - Complete irradiation of the AGR-2 experiment for approximately 550 EFPDs, which will test TRISO particle fuel made at prototypic production scale.

- Complete irradiation of the AGR-3/4 experiments for approximately 350 EFPDs which will contain designed-to-fail (DTF) TRISO particles that are expected to fail during irradiation and collect data on fission product transport.
- Complete the irradiation of the AGR-5/6/7 experiments for approximately 500 EFPDs, which will serve as a fuel qualification test and a margin test based upon the selected fuel fabrication specifications.

Irradiation of the AGR-8 experiment originally conceived as a fission product transport validation test has been deferred at this time because of the lack of a selected reactor design, reduced funding levels, and schedule considerations.

- Safety Testing and PIE:
  - Perform safety testing and PIE of UCO TRISO particle fuel produced at laboratory scale (AGR-1).
  - Perform safety testing and PIE of both UCO and UO_2 TRISO particle fuel from prototypic production-scale equipment to obtain normal operation and potential accident condition performance data (AGR-2).
  - Perform PIE of irradiation capsule components and representative UCO TRISO particle fuel containing DTF particles in support of fission product transport model development (AGR-3/4).
  - Perform safety testing and PIE of the qualification test fuel to demonstrate that the reference fuel meets HTGR fuel performance requirements for normal operating conditions and potential accident conditions (AGR-5/6) and to obtain data needed for assessing the fuel performance margin to failure (AGR-7).
- Fuel Performance Modeling:
  - Improve the existing coated particle material property database to help develop constitutive relations that describe the thermomechanical, thermophysical, and physiochemical behavior of coated particles.
  - Develop a mechanistic fuel performance model for normal and off-normal HTGR conditions and benchmark against relevant performance data.
- Fission Product Transport and Source Term Determination:
  - Evaluate data from irradiation and safety testing of DTF fuel to characterize fission product release and transport from TRISO particle fuel into a fuel compact matrix and fuel element graphite under normal and off-normal HTGR conditions (AGR-3/4).

Understanding the relationships among the fuel fabrication process, fuel product properties, and in-reactor fuel performance is necessary. Fuel performance modeling is also addressed. The performance model is essential for several reasons, including guiding the future plant designer in establishing the core design and operating limits and in demonstrating to the licensing authority that the applicant has a thorough understanding of the in-service behavior of the fuel system and extrapolation of test results.

Irradiation and safety testing activities will also establish the operating margins for the fuel. For HTGR fuel, this means measuring the fuel performance at a combination of temperature, fast neutron exposure, and burnup levels at which the fuel begins to fail and release fission products in significant quantities, either during normal operation or under potential accident conditions. The AGR-7 experiment in irradiation test train AGR-5/6/7 is designed so that some measurable level of fuel failure and/or fission product release is expected to occur.

Opportunities for collaboration have occurred as others in the international community continue developing fuel for an HTGR. A Very High Temperature Reactor (VHTR) Fuel and Fuel Cycle Project

Management Board has been established under the Generation IV International Forum (GIF) VHTR project to identify areas of possible collaboration; some activities are underway. These are mentioned in sections of the document below.

1.2 Program Status

As of May 30, 2020, the AGR Fuel Program had completed the following major tasks:

- Fuel Fabrication
  - Utilized publicly available German coating process information and German fuel and material property data for development of a fuel design and fuel specifications.
  - Used German coating process information in conjunction with coating process information from the U.S. MHTGR and NPR programs to establish a reference set of coating process parameters for laboratory-scale equipment, and verified that these coating parameters yield properties in the prismatic HTGR particle design that are equivalent to the German coating properties.
  - Developed a German-like laboratory-scale overcoating process and a laboratory-scale compacting process.
  - Improved on previous U.S. UCO fuel kernel fabrication methods (forming, calcining, carbothermic reduction, and sintering) that resulted in better carbon dispersion, kernel microstructure, and surface topography.
  - Designed a prototypic production-scale furnace retort and gas distributor nozzle for chemical vapor deposition of the TRISO coating layers and developed parameters that increased the charge mass about thirty-fold relative to the laboratory-scale coater.
  - Re-established basic QC capability for coated particle fuel and developed new QC methods (as required) for enhanced characterization of kernels, coatings, and compacts.
  - Identified an alternate means of producing resinated graphite (matrix) powder by dry jet milling of co-mingled components, thus eliminating methanol as a part of matrix production and reducing preparation time from days to hours. Demonstrated the acquisition of resinated graphite powder as a subcontracted commodity.
  - Identified a pharmaceutical-industry-developed process for overcoating TRISO fuel particles using a resinated (thermosetting) graphite powder, substituting water for methanol as the wetting agent, eliminating the potential generation of a Resource Conservation and Recovery Act (RCRA) hazardous mixed waste. Eliminated particle upgrading, recycle, and reclamation process steps and reduced cycle time from days to a couple of hours.
  - Developed an automated, multi-cavity compacting system with a volumetric feed system that is readily scalable for production.
  - Developed the thermal treatment schedule for compacts and demonstrated a combined cycle furnace for compact carbonization and heat treatment that produces compacts with excellent structure and high matrix density for very good thermal conductivity.
  - Produced and characterized initial reference fuel particles and selected variants for shakedown irradiation testing (AGR-1).
  - Updated reference fuel with a prototypic production-scale coater for fuel performance testing for UCO and UO_2 kernels (AGR-2).

- Produced compacts containing driver and DTF fuel particles for fission product transport testing (AGR-3/4).
- Produced low-enriched UCO kernels, TRISO particles, and fuel compacts for the AGR-5/6/7 experiments.
- Dispositioned all equipment, feedstock materials, and chemicals procured for the fabrication of AGR-5/6/7 fuel.
- Shipped samples of AGR kernels, TRISO particles, and compacts for archival storage at INL.

- Fuels and Materials Irradiations:
  - Completed irradiation of the AGR-1 experiment compacts for 620 EFPDs to a maximum burnup of 19.6% FIMA in the ATR at INL with no fuel particle failures. The AGR-1 experiment was the shakedown test for irradiation, safety testing, and PIE of the initial reference fuel and selected variants from laboratory-scale equipment. The fuel used in the AGR-1 experiment was 19.8% enriched.
  - Completed refurbishment of the dry transfer cubicle at ATR for sizing of the AGR test trains in preparation for shipment from ATR.
  - Completed transport of the AGR-1 test train from ATR to the Materials and Fuels Complex (MFC) at INL to begin PIE in March 2010.
  - Collected, analyzed, and qualified millions of data points generated during AGR-1, AGR-2, and AGR-3/4 fuel fabrication, irradiation, and PIE for future support of Nuclear Regulatory Commission (NRC) licensing activities for TRISO particle fuel.
  - Completed irradiation of the AGR-2 experiment compacts for 559 EFPDs in ATR to a maximum burnup of 13.15% FIMA containing prototypic production-scale UCO and UO_2 fuel with no apparent fuel particle failures. The fuel used in the AGR-2 experiment was 14.0% enriched.
  - Completed shipment of the AGR-2 test train to MFC at INL to begin PIE in July 2014.
  - Completed irradiation of the AGR-3/4 test train in ATR with DTF fuel particles in April 2014 after 369 EFPDs of irradiation to a maximum burnup of 15.27%. The fuel used in the AGR-3/4 experiment was the same as that used in AGR-1 with 19.8% enrichment.
  - Completed transport of the AGR-3/4 test train from ATR to MFC in two shipments in the spring of 2015. Because of the size of the test train, it had to be cut into two pieces to fit into the shipping cask.
  - Initiated the AGR-5/6/7 irradiation in ATR in February 2018; achieved 355 EFPDs as of May 31, 2020.
  - Completed and issued AGR-1, AGR-2, and AGR-3/4 Nuclear Data Management and Analysis System irradiation data qualification reports.
  - Completed and issued the AGR-1, AGR-2, AGR-3/4 as-run final irradiation reports.[12,13,14]
  - Completed and issued the AGR-1 and AGR-2 safety test predictions reports.[15,16]
  - Completed and issued the *Uncertainty Quantification of Calculated Temperatures for AGR-3/4 Experiment*.[17]

- Safety Testing and PIE:
  - Completed PIE and Safety Testing of the AGR-1 irradiated fuel and components, including inert-atmosphere safety tests on 19 compacts, most of which were isothermal tests at temperatures of 1600, 1700, or 1800°C for approximately 300 hours each. In one of the tests, three compacts were simultaneously tested using a varying temperature profile (minimum temperature 830°C, maximum temperature 1690°C) simulating a temperature transient during a core conduction cool-down event.
  - Isolated and studied local SiC degradation responsible for SiC failure and Cs release in AGR-1 UCO, AGR-2 UCO, and UO_2 particles.
  - Completed AGR-1 compact cross section ceramography for evaluating TRISO layer post-irradiation morphology (cracks, tears, inter-layer bonding, etc.)
  - Completed AGR-1 loose-particle ceramography for evaluating kernel swelling and buffer densification.
  - Completed advanced electron microscopy and micro-analysis of as-fabricated, irradiated, and post-irradiation safety-tested AGR-1 particles at INL.[18] Study focused on fission product transport phenomena (e.g., fission product precipitate compositions and distributions in TRISO layers). Issued final report on this work.
  - Prepared and issued the AGR-1 PIE final report,[19] summarizing the findings of the AGR-1 PIE and Safety Testing efforts performed at INL and ORNL.
  - Completed disassembly of the AGR-2 test train and capsules, as well as metrology and nondestructive gamma scanning of compacts and capsule components.
  - Initiated destructive PIE of thirteen as-irradiated AGR-2 compacts, including deconsolidation-leach-burn-leach analysis, follow-on particle exams on nine compacts (eight UCO and one UO_2), and cross section analysis of four compacts (three UCO and one UO_2).
  - Completed eight 1600°C safety tests at ORNL of AGR-2 compacts (two UO_2 and six UCO). Completed one 1500°C test of an AGR-2 UO_2 compact. Completed one 1700°C safety test of an AGR-2 UO_2 compact. Completed five 1800°C safety tests of AGR-2 UCO compacts.
  - Completed a temperature-transient safety test of three AGR-2 UCO compacts. The compacts were simultaneously heated under a varying temperature (minimum temperature 830°C, maximum temperature 1690°C) simulating a temperature transient during a core conduction cool-down event.
  - Completed AGR-2 compact ceramography for evaluating TRISO layer post-irradiation morphology (cracks, tears, inter-layer bonding, etc.).
  - Established contracts to perform PIE and Safety Testing on South African (PBMR) UO_2 compacts from AGR-2 Capsule 4. Four PBMR compacts were safety tested, and one was subject to as-irradiated destructive exams.
  - Completed ceramography, optical microscopy, and dimensional analyses of the kernels and TRISO coatings of about 500 AGR-2 particles from three different UCO compacts and one UO_2 compact. From these exams, the irradiation-induced kernel swelling and buffer shrinkage was assessed.[20]

- Completed burnup evaluation via destructive analyses of two AGR-2 UCO compacts and two UO_2 compacts.[21]
- Completed two complementary journal articles on the effects of irradiation damage on fission product transport within the SiC layers of AGR-1 and AGR-2 TRISO particles. The thermodynamic simulations were applied to help understand the observations from advanced microscopic techniques.[22,23]
- Completed a study on the separate effects testing of high-temperature oxidation of graphitic matrix material in moisture atmospheres.[24]
- Completed disassembly of the AGR-3/4 test train and capsules, as well as metrology and nondestructive gamma scanning of the fuel compacts and capsule components.
- Issued AGR-3/4 first-look report[25] detailing capsule disassembly and component metrology.
- Revised AGR-3/4 as-run thermal analysis[26] based on results of component metrology.
- Completed physical sampling of inner and outer rings from six AGR-3/4 capsules to support measurement of radial fission product profiles in the rings.
- Completed ceramographic exams on three AGR-3/4 compacts to establish the morphology of the designed-to-fail particles and the TRISO-coated driver particles.[27]
- Installed necessary equipment in radiation hot cell and completed radial deconsolidation of eight as-irradiated AGR-3/4 compacts and three compacts that were subjected to heating tests.
- Completed FACS heating tests of four as-irradiated AGR-3/4 compacts.
- Established capability to re-irradiate loose particles and AGR-3/4 compacts in the INL NRAD reactor. Together with subsequent safety testing of the re-irradiated fuel, this enables the study of I-131 release from AGR fuels, a critical part of the fuel safety analysis.
- Completed a total of four separate NRAD reirradiation/FACS tests of the following: four AGR-2 Compact 6-4-1 kernels and 12 intentionally-cracked AGR-2 Compact 5-4-2 particles. Four kernels or particles were used in each test.
- Completed NRAD reirradiation and FACS heating tests of three AGR-3/4 compacts
- Completed final design reviews, initiated equipment procurement, received the custom furnace, all major out-of-cell and in-cell equipment, began Phase I assembly/construction of the Air/Moisture Ingress Experiment (AMIX) for testing irradiated fuels under air/moisture-ingress conditions at INL.
- Completed construction and initiated bench-top testing of furnace and related systems to support development of the hot-cell air/moisture-ingress furnace being designed at INL.

• Fuel Performance Modeling:

- Developed the fuel performance modeling code, Particle Fuel Model (PARFUME), which has been used for irradiation pre-test and safety test predictions
- Published two journal articles comparing the experimental results of AGR-1 PIE[28] and AGR-1 safety tests with PARFUME pre-test predictions.[29]
- Performed pre-test predictions with PARFUME for AGR-2, AGR-3/4, and AGR-5/6/7 along with safety test predictions for AGR-2.

- Published a journal article evaluating the design parameters for TRISO-coated fuel particles to establish manufacturing critical limits using PARFUME.[30]
- Revised the PARFUME Theory and Model Basis Report and PARFUME User's Guide to reflect updates to the code.[31,32]
- Published an assessment of material properties used in PARFUME to identify key material properties that impact fuel performance.[33]

- Fission Product Transport and Source Term:
  - Completed hydrogen and tritium permeation measurements in the HTGR candidate high-nickel superalloys Incoloy 800H, Inconel 617, and Haynes 230.
  - Evaluated available data from AGR-3/4, in particular fission product inventories and distributions in the matrix and graphite rings, and compared these to model predictions. This activity is ongoing. It is expected that it will result in refinement of model parameters that influence fission product transport predictions.

1.3 NRC Assessment Status

In 2014, the NRC staff completed its assessment of two previously submitted NGNP licensing white papers that described the AGR Fuel Program and the approach to determining mechanistic source terms, an approach that relied extensively on data being obtained in the AGR Fuel Program.[4,34] The results of the assessment were documented and transmitted to DOE via a letter with two enclosures.[5] The enclosures provided feedback on key licensing issues that are closely tied to the AGR Fuel Program, its approach to fuel development and qualification, and to mechanistic source terms.

In its assessment, the NRC found:

> *In summary, the staff views the proposed high-level approaches to NGNP fuel qualification and mechanistic source terms as generally reasonable. The staff observes that the fuel development and testing activities completed to date in the AGR Fuel Program appear to have been conducted in a rigorous manner and with early results that show promise towards demonstrating much of the desired retention capability of the TRISO particle fuel developed for NGNP. Moreover, the staff believes that the planned scope of activities in the AGR Fuel Program is reasonably complete within the context of pre-prototype fuel testing.*[25]

Regarding fission product transport phenomena and the collection of supporting data in the AGR Fuel Program, the NRC found:

> *The NRC staff's FQ-MST [Fuel Qualification-Mechanistic Source Term] assessment report concludes, with caveats, that DOE/INL's ongoing and planned testing and research activities for NGNP fuel qualification and mechanistic source terms development appears to constitute a reasonable approach to establishing a technical basis for the identification and evaluation of key HTGR fission product transport phenomena and associated uncertainties. The staff expects more information on release and transport phenomena through event-sequence-specific pathways to be developed as DOE/INL's activities in these areas proceed.*[25]

The "caveats" noted in the NRC assessment pertain primarily to the NRC staff's perceived need for fuel surveillance and testing of fuel fabricated in the production fuel facility and taken from the initial core of the prototype HTGR. Examples of more specific caveats are provided from the following staff finding:

> *The staff acknowledges that the AGR Fuel Program includes significant ongoing and planned research efforts to investigate the poorly understood phenomenology of silver and palladium interactions with TRISO coating layers. DOE/INL has stated that these research efforts may include examinations on fuel samples irradiated in the ATR at temperatures significantly above those normally expected during irradiation in an NGNP core. The staff would consider new insights emerging from such investigations in evaluating the potential fuel performance uncertainties associated with the initially unmet need for test data from real-time fuel irradiations in an HTGR neutron spectrum*[25]

Regarding plans to characterize the effects of air and moisture ingress on oxidation of fuel element graphite and matrix materials,[35] the NRC staff noted:

> *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 envelope all LBEs [licensing-basis events] that involve air or moisture ingress in the final NGNP design will be important.* [25]

These significant NRC findings indicate that the AGR Fuel Program is on track to meet its goal of providing a fuel qualification dataset in support of the licensing and operation of an HTGR.

The NRC conducted an extended pre-application review of the MHTGR design from 1985 to 1996 and produced two draft safety evaluation reports in 1989[36] and 1995.[37] Neither report was issued in final form, but they had the same document number, and the later version extensively referenced the earlier version. Thus, while these documents are considered useful background material, they are not considered official guidance from the NRC. In addition, the NRC initiated pre-application reviews of the GT-MHR (2002-2003), the PBMR/Exelon pebble bed design (2001-2002), and the PBMR/Westinghouse pebble bed design (2004-2010). All three reviews were terminated with no formal NRC assessment.

More recently, AGR program staff collaborated with EPRI to prepare a topical report on UCO TRISO fuel performance based on results from the AGR-1 and AGR-2 fuel fabrication, irradiation, PIE, and safety testing activities.[38] This collaboration also included reactor designers and fuel vendors that participated as members of the High-Temperature Reactor Technology Working Group. The report was submitted to the NRC in 2019, and describes the historic basis for the current TRISO particle design and provides AGR data demonstrating the performance of UCO TRISO fuel particles under the range of conditions enveloped by the AGR-1 and AGR-2 irradiations and safety tests. The NRC is currently evaluating the report, and a Safety Evaluation is expected by the end of FY20. This will accelerate licensing of advanced high-temperature reactors by providing NRC concurrence that TRISO particles fabricated to the AGR specifications will have similar performance as demonstrated by the AGR-1 and AGR-2 fuel under similar service conditions.

2. GOALS, ASSUMPTIONS, AND OBJECTIVES

This section presents the overall set of programmatic goals, assumptions, and objectives developed to guide the preparation of this plan. The scope of the technical program plan is divided into five program elements:

1. Fuel fabrication
2. Fuel and materials irradiation
3. Safety testing and PIE
4. Fuel performance modeling
5. Fission product transport and source term.

Detailed goals, assumptions, and objectives developed to guide the planning of each of these program elements are discussed in Section 3. A high-level set of goals, assumptions, and objectives from the perspective of the overall AGR Fuel Program are identified in Subsections 2.1, 2.2, and 2.3.

2.1 Overall Program Goals

The overall goals for the AGR Fuel Program are to:

- Provide a fuel qualification dataset to support licensing and operating a prismatic HTGR. HTGR fuel performance demonstration and qualification compose the longest-duration R&D task required for design and licensing. The fuel is to be demonstrated and qualified for service conditions encompassing expected normal operating and potential accident conditions.
- Support deployment of the HTGR for hydrogen, process heat, and energy production in the U.S. by reducing the market entry risks posed by technical uncertainties associated with fuel production and qualification.
- Use international collaboration mechanisms to extend the value of DOE-NE resources (primarily through GIF VHTR-related activities).
- Support establishment of a domestic TRISO particle fuel manufacturing capability for fabricating demonstration and qualification experiment fuel.
- Improve understanding of the fabrication process, its impact on as-fabricated fuel properties and attributes, and their impacts on in-reactor performance.

Fuel qualification is herein defined as demonstrating the robust performance and efficacy of the reference TRISO particle fuel by producing experimental data and analytical results.

2.2 Overall Program Assumptions

Overall program assumptions are as follows:

- Government and potential industry co-sponsors of the HTGR recognize that a stable, long-term, disciplined, fuel development and qualification effort offers the greatest probability of success.
- Fission product retention in coated particle fuel at the level demonstrated by the German program in the late 1980s (proof test composite EUO 2358-2365) meets the needs of the U.S. program.

- Proposed HTGR designs may impose more demanding service conditions than the German high-temperature reactor (HTR) Modül and require testing of a fuel based on the prismatic HTGR design and the German coating process.
- It is technically feasible to reestablish, at a reasonable cost, a production capability in the U.S. that is equivalent to the German capability.
- A base technology program aimed at reestablishing the capability to fabricate and test fuel, with a follow-on goal of improving the technology to the point where it can support economic deployment of an HTGR, is the lowest-risk approach to achieving the program goals.
- The target peak time-averaged fuel temperature (1250°C) can support HTGR operation at least to the lower end of the anticipated design core outlet helium-coolant temperature range (750 to 800°C).
- Annual DOE funding allocations that are less than those required to support the planned work scope included in the lifecycle baseline (LCB), as shown in Section 4 (see Figures 4, 5, and 6), will impact plans presented here, causing delays to the schedule or reductions in planned work scope.
- Results of the AGR Fuel Program will be responsive to the design data needs of the reactor and fuel vendors and to the NRC’s fundamental licensing analysis data needs.
- Radiologically significant reactivity transients (those capable of compromising fuel integrity) are precluded by design; consequently, fuel performance and fission product release under these conditions need not be experimentally characterized.
- Activities relating to the licensing of a fuel vendor’s product by the NRC Office of Nuclear Reactor Regulation and meeting the NRC mandate of 10 CFR 50, Appendix B,[39] quality assurance (QA) and QC are outside the scope of this program.
- No major programmatic or technical difficulties that could impact the LCB or schedule will be encountered during the fuel development, irradiation testing, or PIE and Safety Testing. The LCB is the established baseline of all activities included in the AGR Fuel Program, including schedule, performance duration, estimated costs, and logic ties.

2.3 Overall Program Objectives

Key objectives for the AGR Fuel Program are delineated below.

- Establish an HTGR TRISO fuel development and qualification program that will:
  - Produce fuel fabrication specifications that meet the anticipated performance requirements of the reactor designer.
  - Prepare a fuel data manual that captures the correlations and uncertainty estimates for fuel performance and fission product transport that are developed under the AGR Fuel Program for this UCO TRISO particle fuel.
  - Develop and qualify TRISO particle fuel (and the associated fuel specification) by generating and presenting statistically sufficient irradiation and PIE data under normal operating conditions, and safety testing data under potential accident conditions, consistent with anticipated design requirements. The relevant fuel qualification information generated in this effort can be used by HTGR fuel vendors in support of HTGR licensing.
  - Establish, through testing and analysis, the performance margin for this fuel form under normal operating and potential accident conditions.

- Enhance understanding of fuel behavior and fission product transport to improve the fuel performance and fission product transport models under normal operation and accident conditions.
- Develop pertinent fuel process information that can be used by HTGR fuel vendors to select and implement fuel fabrication processes.

All activities that have direct input to the irradiation test specimen fabrication, irradiation campaigns, and safety testing will be conducted in accordance with national consensus standard Nuclear Quality Assurance (NQA)-1-2008/1a-2009, "Quality Assurance Requirements for Nuclear Facility Applications,"[40] published by the American Society of Mechanical Engineers. Each participating organization shall prepare specific QA plans for its assigned scope of work and may prepare additional project-specific plans for individual work breakdown structure (WBS) elements as appropriate.

3. PROGRAM ELEMENTS

This section summarizes detailed goals, assumptions, and objectives associated with the individual program elements and the activities performed or required to meet these and the high-level goals and objectives identified in Section 2. Program elements discussed in more detail below include fuel fabrication, fuel and materials irradiation, PIE and Safety Testing, fuel performance modeling, and fission product transport and source term.

3.1 Fuel Fabrication

3.1.1 Goals, Assumptions, and Objectives

The goals, assumptions, and objectives specific to this program element are as follows.

3.1.1.1 Goals

- Establish a production-scale TRISO particle fuel fabrication technology in the U.S. that is capable of producing fuel at a quality level at least equivalent to those of German fuel particles from composite EUO 2358-2365.
- Develop a fundamental understanding of the relationships among fuel fabrication process parameters, fuel product properties, and fuel performance under normal operating and potential accident conditions.
- Develop appropriately automated fuel fabrication technology suitable for mass production of coated particle fuel at an acceptable cost and at acceptable levels of high quality and consistency.
- Develop and document the manufacturing processes required to meet the fuel process and product specifications that will be developed to satisfy Goals 2 and 3 above.

3.1.1.2 Assumptions

- The coated particle design to be qualified in the AGR Fuel Program will be based on the most stringent performance requirements for two different types of HTGRs (pebble-bed and prismatic). This approach will result in the qualification of a fuel performance envelope that can be used by either HTGR technology.
- Fuel capable of acceptable performance up to a target peak time-averaged fuel temperature of 1250°C in normal operation (as well as during associated accident conditions for relevant designs) can support HTGR operation within a substantial portion of the anticipated core outlet helium-coolant temperature range (750 to 800°C).

- The capability to economically mass-produce high-quality, coated particle fuel elements is a prerequisite for commercial viability of HTGRs.
- The low-enriched UO_2 particles qualified by the Germans in pebble-bed reactors for burnup to about 10% FIMA are not adequate for higher fuel burnup (16 to 22% FIMA), higher operating temperatures, and temperature gradient service in prismatic HTGRs.
- Fuel particles made with low-enriched UCO kernels and having coating properties equivalent to those of German fuel particles from composite EUO 2358-2365 (that were irradiated in the HTR-Modül proof tests [HFR-K5 and -K6 in Petten, Netherlands]) with no in-pile failures will perform well in fuel compacts under prismatic HTGR irradiation conditions.
- The lowest-risk path to successful manufacturing of coated fuel particles is to closely replicate the proven German coating technology to the extent possible in a coated fuel particle design, incorporating the lessons learned from U.S. fabrication and irradiation experience to improve the coating process.

3.1.1.3 Objectives

- Support establishment of and demonstrate coated particle fuel fabrication capability from kernel production through fuel compact production.
- Conduct fuel kernel process studies to optimize the UCO kernel fabrication process (carbon dispersion, broth chemistry, calcination, carburization, and sintering).
- Conduct fuel-coating process studies to determine the capability to replicate the properties of German coated particle fuel for HTGR fuel and to establish conditions that yield coating layers with microstructural properties and features comparable to those in the German fuel particles in proof-test composite EUO 2358-2365.
- Develop a process suitable for large-scale fuel production that produces coating properties consistent with acceptable fuel performance. This will be accomplished using a coater that provides a coating environment similar to the German production-scale coater and has appropriate features for a production-scale coater (for loading, unloading, sampling material from the coater, and cleaning).
- Develop additional QC methods to improve fuel characterization capabilities and results.
- Fabricate fuel as needed for irradiation testing, including DTF fuel for fission product transport tests. The fuel shall meet the product requirements specified in the test fuel product specifications. These fuel product specifications will be based on specific objectives for each irradiation experiment.
- Prepare a fuel product specification and process specification for large-scale HTGR fuel fabrication that defines all requirements the fuel must satisfy to ensure acceptable performance under HTGR normal operating and potential accident conditions.
- Develop automation technologies that can be applied to fuel fabrication processes to the maximum extent practicable.

3.1.2 Scope of Fuel Fabrication

The ultimate fabrication goal for HTGR fuel is the economical production of high-quality kernels, TRISO-coated fuel particles, and compacts or pebbles that meet the fuel product specifications. The fuel fabrication activities described herein are intended to develop and qualify a fuel fabrication process that is the foundation for fabrication of production-scale, coated particle fuel for HTGRs. These activities must optimize the process to achieve the required kernel, coated fuel particle, compact, or pebble characteristics and quality. They must also result in scale-up of kernel production, coating, and compact or pebble fabrication processes.

Coated particle fuel fabrication differs from light-water reactor fuel manufacturing. The fabrication process developed within the AGR Fuel Program begins with low-enriched UCO kernels formed by the internal gelation process in which droplets of uranium-containing chemical broth are formed into gel spheres in a fluid medium. The resulting gel spheres are dried and sintered into hard ceramic particles yielding kernels of a controlled, consistent size and chemistry.

Fuel kernels are coated using a fluidized-bed chemical vapor deposition process. The coatings include a low-density carbon (buffer) layer, a high-density inner pyrolytic carbon (IPyC) layer, a SiC layer, and a high-density outer pyrolytic carbon (OPyC) layer. These coatings are designed to work together to make each fuel particle a mini pressure vessel that will maintain its integrity and retain fission products during normal reactor operation and potential accident conditions. The finished coated particle is a small (≤1 mm outside diameter [OD]) carbon and ceramic sphere that is stable to temperatures well beyond 1600°C.

The coated fuel particles are formed and pressed into physical shapes for use in the reactor, as shown in Figure 1. For the prismatic reactor design, fuel particles are pressed into cylindrical-shaped compacts for insertion into large hexagonal graphite blocks, which are stacked in columns to form the reactor core. Fuel particles for the pebble-bed reactor design are pressed into tennis-ball-sized pebbles that may be recirculated in the reactor. For both designs, the particles are overcoated with a carbonaceous matrix composed of graphite powder and a resin binder, formed into the desired shape, carbonized, and treated at high temperature to provide a thermally stable material.

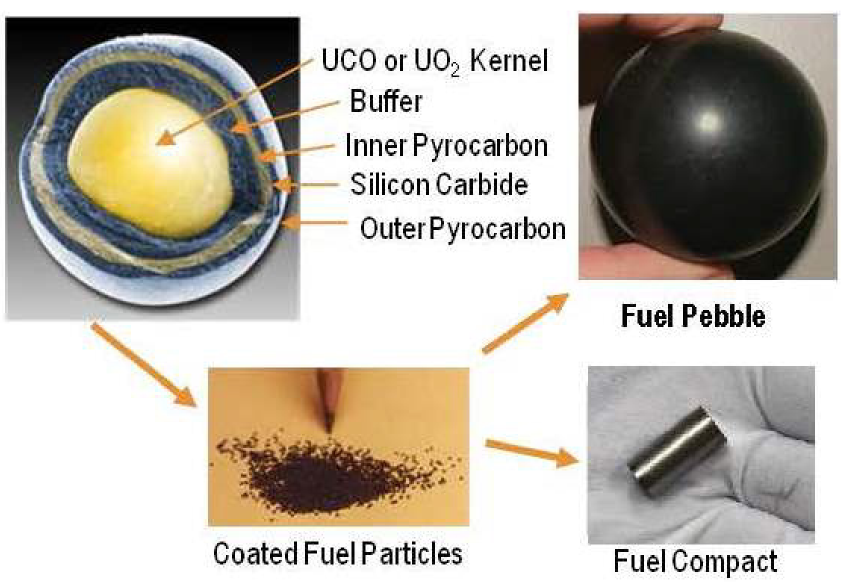

Figure 1. Formation of potential fuel forms.

The target quality level for coated particle fuel is based on the quality level achieved in the German program in the late 1980s, with the EUO 2358–2365 fuel particle composite used in the HTR-Modül proof tests taken as a standard for comparison,[41] in combination with core-design-driven quality specifications derived during the GT-MHR conceptual design.[111] The AGR fuel fabrication effort was designed to expand the understanding of the relationships among kernel and coating properties, fabrication process conditions, and the irradiation performance of the fuel. The earlier U.S. and German manufacturing efforts and subsequent work in other national programs achieved a substantial level of understanding of these relationships, but additional work is required.

Fuel failures in U.S. MHTGR and NPR program irradiation tests have been analyzed[9,10] along with U.S. and German fuel fabrication processes and irradiation performance.[42] These studies suggest key differences between German and historical U.S. coating processes, and that coating properties contribute to better irradiation performance. The most significant differences in the German processes are: (1) a greater deposition rate of pyrocarbon layers, resulting in more isotropic coatings with greater stability to high fast neutron fluence under irradiation; (2) more intimate bonding of the IPyC and SiC coating layers; (3) continuous coating of all layers, resulting in less potential for as-manufactured defects and possible beneficial effects on coating properties; and (4) a lower SiC coating temperature, resulting in a smaller grain size. In addition, the German compacting process began with overcoating the coated particles with a graphite/resin blend to prevent particle-to-particle contact during pressing, versus a pitch-injection process used by earlier U.S. fuel programs. Thus, the starting point for fuel fabrication development was the U.S. kernel and compacting experience coupled with knowledge of the German coating and overcoating processes as supplemented by lessons learned from fuel technology development within the U.S.

The work to produce TRISO particle fuel that meets the specifications has included kernel process development, coating process development, overcoating and compacting process development, advanced characterization and QC methods development, and process documentation. The scope of fuel manufacturing activities is summarized below.

3.1.2.1 Prepare Irradiation Test Fuel Specifications. Developing a fuel fabrication process and fabricating irradiation experiment fuel in a manner that complies with the QA requirements of NQA-1[42] is based on the specifications of kernel, coated particle, and compact properties and on key process parameters. Detailed product specifications, along with a limited set of process specifications affecting microstructure characteristics (which are considered to be important to irradiation performance but cannot be fully characterized) are required for each irradiation experiment conducted within the program. These specifications include the parameters identified in Table 1. (Property specifications include properties for individual batches as well as for composited lots formed from multiple batches.)

Execution of this plan produced specifications for the fuel to be used in the series of AGR irradiation experiments, leading to a specification for fuel to be produced for an HTGR.

Table 1. Fuel specification parameters.

Parameter	Mean	Critical Region[a]	Fraction in Critical Region
Kernel Composite			
^{235}U Enrichment	X		
C/U ratio	X		
O/U ratio	X		
(C+O)/U ratio	X		

Table 1. (continued).

Parameter	Mean	Critical Region[a]	Fraction in Critical Region
Individual Impurities (Li, Na, Ca, V, Cr, Mn, Fe, Co, Ni, Cu, Zn, Al, and Cl)	X		
Process Impurities (P, S)	X		
Envelope Density	X		
Diameter	X	X	X
Aspect Ratio		X	X
Microstructure	Visual Standard		
Coated Particle Composite			
Buffer Density[b]	X		
Thickness (Buffer, IPyC, SiC, OPyC)	X	X	X
Density (IPyC[b], SiC, OPyC)	X	X	X
Anisotropy (IPyC, OPyC)	X	X	X
SiC Aspect Ratio		X	X
Defective IPyC Fraction	X		
Defective SiC Fraction[d]	X		
Defective OPyC	X		
Pre-Burn Exposed Uranium	Measurement Only		
Post-Burn Exposed Uranium	Measurement Only		
SiC Soot Inclusions[c]	Measurement Only		
SiC Microstructure	Visual Standard		
Heat-Treated Compacts			
Uranium Loading	X		
Diameter		X	X
Length		X	X
Matrix Density	X		
Impurity Content (Fe, Cr, Mn, Co, Ni)	X	X	X
Impurity Content (Ca, Al, Ti, V)	X		
Heavy Metal Contamination Fraction[d]	X		X
Exposed Kernel Fraction[e]	X		X
Dispersed Uranium Fraction[e]	X		
Defective SiC Fraction	X		
Defective OPyC Fraction	X		

a. The specification of a critical region boundary and the fraction of particles within the critical region are provided to limit the distribution tail of a property or, in the case of attribute properties, the subpopulation of abnormal particles with a specific defect.
b. Calculated from pooled characterization data for similar particle batches when direct measurements were unavailable.
c. An indication of defects within the SiC layer, historically identified by General Atomics as “Gold Spots,” but not detectible as such in the more opaque, finer-grained SiC shells.
d. AGR-1 and AGR-2 experiment only.
e. AGR-5/6/7 experiments only.

3.1.2.2 Fuel Kernel Manufacturing. As discussed in Section 1, the AGR Fuel Program elected to develop coated particle fuel using a low-enriched UCO kernel to support the NGNP project. Low-enriched UCO kernels had been produced for earlier irradiation testing in the U.S. by General Atomics (GA)[43,44] and BWX Technologies Inc. (BWXT-NOG).[45] Currently, BWXT-NOG possesses the only commercial domestic fuel fabrication facilities that can handle uranium enrichment levels in excess of 5%. The BWXT-NOG internal gelation low-enriched UCO kernel production process was selected for the AGR Fuel Program with the understanding that additional process development would be needed to improve the overall quality of the product and adjust for the kernel diameters specified. Thus, the scope of fuel kernel manufacturing included the following elements.

- Process development:
  - Achieve specified kernel density
  - Improve carbon dispersion in the acid-deficient uranyl nitrate solution used in kernel formation
  - Optimize the sintering process
  - Reduce process variability
- Produce natural UCO kernels to support coating process development
- Produce low-enriched UCO kernels for use in production of fuels to be irradiated.

3.1.2.3 Coating Process Development. When the AGR Fuel Program began, no active coating process facilities existed within the U.S. The GA coater used to coat the fuel irradiated in High Flux Isotope Reactor (HFIR) Removable Beryllium-21 (HRB-21) and NPR capsules had been shut down for more than a decade, and the facility had been completely dismantled. Small coaters remained at ORNL and BWXT-NOG, but neither had been operational for the production of TRISO fuel for many years. Thus, a U.S. TRISO particle coating capability needed to be reestablished. In addition, root cause assessments of the HRB-21 and NPR capsule fuel particle failures[9,10,42,] indicated a need to adjust the coating process parameters to change the IPyC, SiC, and OPyC layer microstructures. Given the successful performance of pyrolytic carbon (PyC) and SiC coatings produced by the German program, a primary objective was to identify process parameters that would produce coating characteristics equivalent to German coatings. Another objective was developing an improved understanding of the relationship between coating process parameters and key coating characteristics known to be important for irradiation performance.

A relatively large number of coating runs was required during initial development to obtain process conditions and durations that produced the desired coating properties and thicknesses. These runs were conducted in a laboratory-scale coater to limit the cost and quantity of materials required, as well as to minimize waste. The assessments noted above concluded that a focus on limiting uranium dispersion during application of the SiC layer by reducing permeability of the IPyC layer resulted in an IPyC layer that was prone to failure during irradiation. Thus, the process development scope included a study of the relationships among IPyC coating conditions, IPyC layer permeability, and IPyC properties that influenced irradiation performance (density, anisotropy, and surface-connected porosity).

Laboratory-scale coater runs established the process conditions needed to produce particles that met the specifications and improved understanding of the relationship between process parameters and key coating properties, but some uncertainties regarding the relationship between properties and irradiation performance remained. Therefore, a reference fuel specification and variations in key coating parameters were needed to provide confidence in achieving acceptable performance in the first irradiation experiment (AGR-1). Use of multiple coated particle types (a baseline and three variants) while meeting the AGR

Fuel Program schedule and funding constraints required that the fuel for the first irradiation (AGR-1) and initial fission product transport irradiations (AGR-3/4) be produced in the ORNL laboratory-scale coater.

Producing the quantities of fuel required to support initial HTGR operation (first core) required a larger coater, so fuel qualification based on fuel produced in a larger coater was a goal. Thus, coater scale-up issues needed to be addressed in the context of defining the coater size and configuration for producing the particles used in subsequent irradiation tests and producing fuel for the initial HTGR. A 6-in.-diameter coater was selected as the “large” size for the coating scale-up effort. Although a larger coater (or multiple 6-in. coaters) would likely be needed for large-scale commercial fuel manufacturing to support deployment of multiple HTGR plants, a 6-in.-diameter coater was selected as a reasonable size for the initial scale-up effort because it was considered adequate for production of fuel for a first-of-a-kind HTGR. The results of the small coater operation were used to reduce the number of large coater runs needed to achieve the specified coating properties, but process development scope was needed to define the large coater process conditions. The large coater was then used to produce the coated particles needed for the AGR-2 irradiation experiment and subsequently used to produce the fuel needed for the AGR-5/6/7 irradiation experiments.

3.1.2.4 Compacting Process Development. Historically, U.S. compacting technology has used a thermoplastic matrix consisting of petroleum pitch mixed with graphite powder and injected into a mold containing fuel particles to make compacts. The injection process can result in high stresses on the particles where point-to-point contact occurs, which is a potential mechanism for particle failure. The compacts were also packed in alumina powder during carbonization to prevent them from losing their shape. The raw materials used in the thermoplastic matrix had relatively high concentrations of metallic impurities that were highly reactive with SiC at high temperatures. The alumina powder used in the carbonization process was another source of impurities that potentially attacked the SiC layer.

Shortcomings in the historical U.S. compacting process were addressed during AGR-1 laboratory-scale compact development by using purified graphite and resin material and a German-like overcoating process to prevent particle-to-particle contact during pressing. The selected thermoplastic resin was similar to one of the resins used successfully by the German program; this eliminated the need for compact support during carbonization.

A thermosetting resin-based matrix process was selected for production-scale fuel manufacturing. This thermosetting resin-based matrix was also formulated from raw materials having low levels of impurities, and it yields stronger, less friable compacts. The thermosetting matrix process can also involve lower compacting forces, thereby reducing the potential for damage while allowing for increased matrix density.

Compacting process development scope included:

- Replicating the matrix formulations of a German thermosetting resin/graphite blends
- Jet-milling the resin, graphite, and hexamethylenetetramine mixture to provide a very uniform matrix supply without the use of methanol to solvate the resin
- Substituting water for methanol during TRISO particle overcoating with the intention of eliminating the generation of a RCRA mixed hazardous waste and pursuing necessary development of a waste disposition path
- Demonstrating prototypic production-scale overcoating equipment and establishing process conditions needed to uniformly overcoat particles with the matrix

- Demonstrating automated pressing equipment and establishing process conditions needed to form the overcoated particles into compacts and performing carbonization and final heat treatment in a furnace capable of combining the two steps
- Producing compacts needed for characterization and irradiation in the AGR Fuel Program's final AGR-5/6/7 irradiation test.

3.1.2.5 QC Methods Development and Application. QC methods were needed to demonstrate that the fuel fabricated for the AGR Fuel Program complied with the product specifications. As with the coating process, facilities were unavailable to measure the properties identified in Engineering Design File (EDF)-4380, "AGR-1 Fuel Product Specification and Characterization Guidance,"[46] at the required confidence levels (typically 95% confidence). Therefore, the development of QC methods involved re-establishing traditional characterization procedures at ORNL and BWXT-NOG and developing advanced QC methods, mainly at ORNL. The following QC capabilities were needed for the inspection and testing of kernels, TRISO particle fuel, and compacts to demonstrate compliance with fuel product specifications:

- Chemistry of kernel batches and composites (carbon, oxygen, uranium, and 15 impurities)
- Kernel ^{235}U enrichment
- Ceramography to provide images for coated particle analysis
- Hardware and software upgrades to the ORNL ellipsometer (2-MGEM) to measure PyC anisotropies
- Automated image analysis for kernel and particle diameter, aspect ratio, and coating thickness measurements
- Density gradient columns for PyC and SiC sink-float density measurements
- Mercury porosimetry for measuring kernel and buffer envelope density and for PyC surface-connected porosity measurements
- An improved technique for measuring PyC coating anisotropy
- Compact measurements, including length, diameter, mean uranium loading, total mass, matrix density, and defective OPyC coating fractions
- Deconsolidation and leach-burn-leach (LBL) testing of fuel compacts to determine the exposed kernel, dispersed uranium, and defective SiC fractions, as well as the quantity of specified impurities outside the SiC layer
- X-ray analysis for detecting uranium dispersion in coated particles as a metric for defective IPyC
- Inspection of particles for soot inclusions and other abnormalities in the SiC layer
- X-ray analysis for detecting gross soot inclusions and misshapen particles in addition to defective IPyC
- X-ray tomography for improved characterization of the internal structure of unirradiated and irradiated fuel particles.

3.1.2.6 Fuel Product and Process Documentation. The description of fuel fabrication development, irradiation, PIE, and safety testing in this plan, when combined with additional reactor design information, provides the information to finalize the top-tier fuel product specifications that define requirements for fuel to be used in an HTGR. Additional reports will be produced to document process and QC development as well as pre- and post-irradiation data for all irradiation tests. The process development and product data compilation reports will provide a basis for the final process parameters necessary to fabricate fuel that consistently meets the fuel product specifications and performance requirements of an HTGR, and the allowable process variations (to the extent determined by the process development tasks).

3.2 Fuel and Materials Irradiation

Irradiation testing of coated particle fuels occurred routinely in the U.S. from the 1960s through the early 1990s. Materials test reactors are still in operation, and personnel experienced with all aspects of irradiation test train design, assembly, and monitoring are active at INL and ORNL. ORNL irradiated fuel for the MHTGR,[47] and both laboratories were involved in irradiation testing of NPR and MHTGR fuel in the early 1990s. ATR at INL and HFIR at ORNL are capable of irradiation testing of AGR fuels. ATR was selected in large part because of the availability of an irradiation location that has a very close match to the nominal gas reactor conditions, resulting in an excellent approximation of HTGR burnup and fast fluence.

3.2.1 Goals, Assumptions, and Objectives

The goals, assumptions, and objectives of fuel and materials irradiation are as follows.

3.2.1.1 Goals

- Provide data for fuel performance during irradiation to support fuel process development, qualify fuel for normal operating conditions, and support development and validation of fuel performance and fission product transport models and codes.
- Provide irradiated fuel and materials for PIE and Safety Testing.

3.2.1.2 Assumptions

- Accelerated irradiation in ATR (up to a maximum of three times real time in terms of both power and fast flux) is equivalent to or is conservative relative to real-time irradiation.
- Developmental fuel fabrication capability is established to provide fuel samples for near-term irradiation.
- Limited material sample irradiations can be conducted in conjunction with fuel irradiation without requiring additional test trains.
- Radiologically significant reactivity transients are precluded by inherent characteristics of the design, so no reactivity insertion accident testing is planned.
- Fuel fabrication capability is established to provide fuel samples representative of high-volume production for qualification testing.
- Waste activated/contaminated metal (lead-out, gas lines, thermocouple [TC] leads, etc.) will be staged in the ATR canal until a cleanup campaign is conducted by ATR Operations. There is no additional cost to the AGR Fuel Program for disposal of this waste.

3.2.1.3 Objectives

- Establish the range of irradiation conditions (power, burnup, flux, fluence, temperature, and environment) based on the needs of the reactor designs and the needs of the associated topical report licensing strategy to qualify fuel for normal operation.

- Establish allowed tolerances on control of irradiation conditions.
- Complete design and fabrication of test trains for irradiation testing of TRISO particle fuel.
- Establish and conduct a fuel and materials irradiation activity that will provide:
  - Independently controlled and monitored capsules within an irradiation test train.
  - Control capability to maintain conditions within the planned tolerances.
  - Online monitoring of release of indicator fission product gases such as krypton and xenon isotopes.
  - A test train design that will allow post-irradiation measurement of metallic fission product release, such as silver (Ag), Cs, and strontium (Sr), from fuel in each capsule during irradiation.
  - Sufficient data to qualify the fuel for normal operation over the required range of irradiation conditions and to support code and model development.
  - Irradiated fuel and material specimens required to support PIE, post-irradiation phenomenological testing, and safety testing activities.

3.2.2 Scope of Fuel and Materials Irradiation

In producing the original version of the technical program plan, the fuel irradiation working group developed a description of the tasks associated with irradiation testing of a representative test train in ATR. Even though the details of test train internals, test articles, and control parameters will vary depending on the requirements for a given irradiation, as defined in the applicable experiment specification, the basic tasks remain the same. This task list, along with corresponding deliverables and interfaces with other activities, has served as the basis for schedule and cost estimates for irradiation testing. The following tasks were identified:

1. *Experiment specification*. This task will specify the test articles, irradiation conditions, and results needed to support fuel fabrication, model development, and plant design and licensing. The experiment specification document will include a definition of test articles to be included in the test train, required operating conditions (including tolerances), and required data (including accuracies) to be produced by the experiment.
2. *Test train and supporting systems' technical and functional requirements*. This task will establish the detailed requirements necessary to proceed with the test train and supporting systems' design in accordance with the experiment specification. The resulting document will include general design requirements associated with the service conditions of the test train in the reactor, design and functional requirements specific to the test train and its supporting systems, and provisions for QA. The document will also include the requirements placed on the experiment by ATR necessary to meet ATR technical specifications and safety analysis report requirements (materials allowed, departure from nucleate boiling ratio, flow instability ratio, etc.).
3. *Test train and supporting systems' design*. This task will establish the detailed design and procurement specifications necessary to proceed with test train fabrication and assembly and establish the needed supporting systems for either a new test train design or replication of a proven test train design.
4. *Test train and supporting systems' fabrication/assembly*. This task includes procuring or fabricating test train components in accordance with the specifications; installation of the

components; refurbishment of supporting systems, as necessary; and assembly of the test train, including the test articles, so that it is ready for insertion into the reactor.

5. *Approval of test articles*. This task includes the receipt, inspection, and QA acceptance of all test articles (compacts, pebbles, loose particles, and/or material samples) to be incorporated into the test train.

6. *Review/approval of final design and fabrication data packages*. This task includes review and concurrence by affected program participants.

7. *Irradiation*. This task addresses all activities associated with irradiation of the test train, including preparation of detailed operating procedures for test train handling during insertion and removal, preparation of experiment safety analysis documentation, preparation of the experiment safety assurance package, test train insertion into and removal from the reactor, operation of the fission product monitoring system, technical support, operation of data-acquisition systems, documentation of conditions and results of irradiation (including a near real-time remote data acquisition capability), and placement of the test train in the ATR canal for cooldown once irradiation is completed.

8. *Cooldown and preparation for shipping*. This task addresses storage (estimated to be about 90 days) of the test train in the ATR canal until the decay heat and radiation levels (from fuel and activated metal) are sufficiently low to proceed with sizing of the test train in the dry transfer cubicle at ATR. Preparations include development of mockups of the test train, development of detailed operating procedures for the sizing activity, dry runs of the planned sizing evolution, actual sizing of the test train, and loading of the shipping cask or package for shipment of the test train to MFC for PIE and Safety Testing. A GE-2000 cask was leased for shipment of the AGR-1, AGR-2, and AGR-3/4 test trains. The AGR-5/6/7 test train may be shipped in an alternative shipping package. The size of the test train will determine the shipping configuration and number of shipments required.

9. *Waste disposition*. The lead-out and test train cuttings are waste forms associated with the AGR irradiations. The lower non-fueled section of the test train is cut off in the ATR canal and temporarily disposed of there as waste. The upper non-fueled section of the test train is cut off in the dry transfer cubicle and then removed and placed in the ATR canal as waste. The cuttings from the test train sizing evolutions are captured in a tray and placed in the ATR canal as waste. These waste sections are dispositioned with other activated/contaminated metal during the course of routine cleanup activities of the ATR canal. The test train gas lines and thermocouple leads are left in the lead-out and dispositioned at the same time.

3.2.3 AGR Irradiations

The number and type of test trains to be irradiated were planned based on the needs of the fuel manufacturing, fuel performance modeling, and fission product transport activities. The selected test train concept used in the first two irradiations, AGR-1 and AGR-2, were placed in large B positions of ATR. The AGR-1 “shakedown” test train contained six capsules independently controlled for temperature and separately monitored for fission product gas release, with each capsule containing twelve 1-in.-long by ½-in.-diameter compacts. The AGR-2 test train contained six capsules independently controlled for temperature and separately monitored for fission product gas release. The U.S.-made UCO fuel was included in three capsules, and UO_2 fuel was included in one capsule. The fifth capsule contained French UO_2 fuel, while the sixth capsule contained South African UO_2 fuel. To increase the capacity for irradiation of fuel and decrease the duration of its irradiation, the AGR-3/4 test train was designed for the

ATR northeast flux trap (NEFT) position. The AGR-3/4 test train contained 12 capsules, with each capsule containing four ½-in.-long by ½-in.-diameter compacts. The AGR-3/4 test train capsules were independently controlled for temperature and separately monitored for fission product gas release. The design and configuration for the AGR-5/6/7 experiments consists of five capsules of varying lengths containing 1-in.-long by ½-in.-diameter compacts with irradiation planned in the NEFT.

The B positions in ATR are located in four triangular arrays, with each array comprising two small B positions and one large B position. The arrow labeled “Small B Position” in Figure 2 points to one of the eight small B positions, which are adjacent to the driver fuel. The arrow labeled “Large B Position” points to the large B position in the east quadrant of the core. This B position is one of four located further from the driver fuel in the beryllium reflector that have a higher ratio of thermal-to-fast flux. Reactor physics calculations conducted by INL for the large B positions show a ratio of burnup to fast fluence that is well matched to expected HTGR conditions. The physics calculations are refined when the actual fuel loadings are known for each test train. The AGR-1 and AGR-2 experiment irradiations were scheduled for about three years depending on the ATR operating schedule to reach the planned approximately 600 EFPDs of irradiation. In actuality, the AGR-1 experiment was irradiated for 620 EFPDs starting in December 2006 and ending in November 2009, and the AGR-2 experiment was irradiated for 559 EFPDs starting in June 2010 and ending in October 2013. The AGR-3/4 experiments completed irradiation in the NEFT in April 2014 after 369 EFPDs, having initiated irradiation in December 2011 and having reached the target burnup levels for all capsules. The NEFT irradiation position used for the AGR-3/4 test train (arrow at top upper right side of Figure 2) can accommodate larger test trains at increased power levels to reduce irradiation times. The NEFT is also being used for the AGR-5/6/7 experiments. Preliminary calculations indicate irradiation times on the order of about three years depending on the ATR operating schedule in the NEFT location to reach the planned 500 to 550 EFPDs of irradiation needed to achieve targeted burnup for the AGR-5/6/7 experiments.

Continuous gas monitoring capability for the AGR-1 and AGR-2 experiment capsules within the test train was provided by a set of six dedicated fission product monitors plus one online operating spare. For the AGR-3/4 experiments irradiated in the NEFT, continuous gas monitoring was provided by 12 dedicated fission product monitors plus two online operating spares. The AGR-5/6/7 experiments are designed for five capsules, and continuous gas monitoring is provided by five dedicated fission product monitors plus two online operating spares.

The seven experiments were identified based on discussions among the working groups during development of the original plan. Program budget constraints and further development of the test train designs have altered the type of test trains initially planned to be used for individual irradiations. For example, it was decided to conduct the AGR-3/4 and AGR-5/6/7 irradiation testing in the NEFT within ATR; the NEFT accommodates a larger test train (and therefore a greater number of fuel compacts) and has a higher acceleration factor to shorten the irradiation schedule timeframe.

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

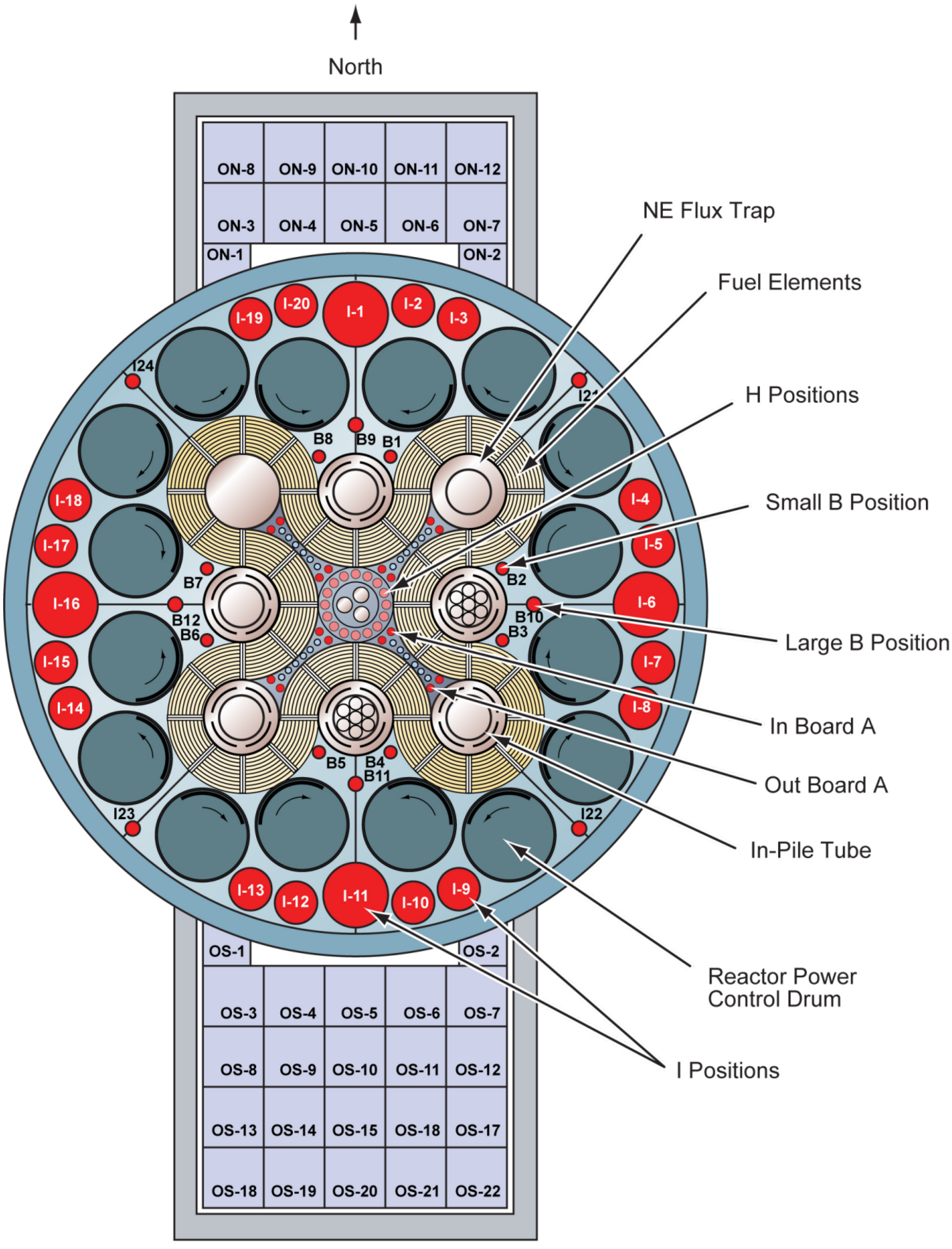

Figure 2. ATR cross section.

3.2.3.2 Performance Test Fuel Experiment (AGR-2). This multi-monitored capsule test train included three capsules of 12 compacts (fabricated at laboratory scale using the same process conditions as AGR-1), each containing U.S. UCO particles made in a production-scale 6-in. coater using process conditions derived from the production of AGR-1 Variant 3 (SiC layer produced using a mixture of hydrogen and argon diluent gases). The UCO compacts were subjected to a range of burnups and temperatures exceeding anticipated reactor service conditions in all three capsules. The test train also included three additional capsules of six to 12 compacts, each containing UO_2 particles produced independently by three program participants (BWXT-NOG, Westinghouse/Pebble Bed Modular Reactor SOC Ltd., and Commissariat a l'Energie Atomique/AREVA), with UO_2 particles from BWXT-NOG and Pebble Bed Modular Reactor SOC Ltd. also compacted using the AGR-1 laboratory-scale process. The range of burnups and temperatures in these capsules exceeded anticipated pebble-bed reactor service conditions. This test train provided irradiated fuel performance data and irradiated fuel samples for safety testing and PIE for key fuel product and process variants. The data obtained from the AGR-2 irradiation and subsequent PIE and safety testing will further increase the fundamental understanding of the relationship among the fuel fabrication process, as-fabricated fuel properties, normal operation, and potential accident condition performance.

3.2.3.3 Fission Product Transport Experiments (AGR-3/4). This multi-monitored capsule test train was a combination of the AGR-3 and AGR-4 experiments originally planned as separate irradiations in large B positions but were combined and placed in the NEFT position in ATR, as also shown in Figure 2. This test train included compacts containing AGR-1 "driver" fuel particles and seeded with 20 DTF fuel particles, each within rings of graphitic material. DTF fuel particles for use in fission product transport testing consisted of reference kernels with only a ~20-µm-thick pyrocarbon seal coating that was intended to fail as designed during irradiation and provided known fission product source terms. The sweep gas not only contained a mixture of helium and neon necessary to provide thermal control of the experiment but also, in one capsule, gaseous impurities (CO, H_2O) typically found in the primary circuit helium of HTGRs. This allowed assessment of the effect of impurities on intact and DTF fuel performance and subsequent fission product transport. The test train was designed to provide data on fission product diffusivities in fuel kernels, as well as sorptivities and diffusivities in compact matrix and graphite materials for use in upgrading fission product transport models. The AGR-3/4 experiments have also provided irradiated fuel performance data on fission product gas release from failed particles and irradiated fuel samples for safety testing and PIE. The in-pile gas release, PIE, and safety testing data on fission gas and metal release from kernels will be used in developing improved fission product transport models to the extent possible from the experimental results.

3.2.3.4 Fuel Qualification and Fuel Performance Margin Testing Experiments (AGR-5/6/7). This multi-monitored capsule test train is a combination of the AGR-5, AGR-6, and AGR-7 experiments, which were originally planned for separate irradiations in large B positions similar to those for AGR-1 and AGR-2, but have been combined for irradiation in the NEFT position in ATR as shown in Figure 2; this is similar to the AGR-3/4 experiments. The test train includes a single fuel type made using process conditions and product parameters considered to provide the best prospects for successful performance based on process development results and available data[a] from AGR-1 and AGR-2 irradiations. This is the reference fuel design selected for qualification. Variations in capsule conditions (burnup, fast fluence, and temperatures) were established in the irradiation test specifications. The sweep gas will contain helium and neon. This test train will provide irradiated fuel performance data

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

and irradiated fuel samples for safety testing and PIE in a sufficient quantity to demonstrate compliance with statistical performance requirements under normal operating and potential accident conditions.

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

3.3 PIE and Safety Testing

This program element assesses the performance of TRISO particle fuel during irradiation and under potential accident conditions. PIE and safety (heat-up) testing are strongly interwoven, because many of the PIE procedures applied to fuel samples following irradiation are also applied to fuel following safety testing. Fuel performance evaluation focuses on quantifying the level of fission product release from the fuel particles and compacts, and on characterizing the condition of kernels and coatings to determine the effect that irradiation or post-irradiation heat-up has on particle microstructure. This work will support the future fuel manufacturing effort by providing feedback on the performance of kernels, coatings, and compacts under varying conditions. Data from PIE and Safety Testing, in conjunction with in-reactor measurements (primarily fission gas release-rate-to-birth-rate [R/B] ratios), are necessary to demonstrate that the quality and performance of the fuel system meet the reactor design requirements. Thus, data from this activity will likely constitute a primary element of the licensee's fuel qualification submittal to the NRC to obtain an operating license for the first plant.

3.3.1 Goals, Assumptions, and Objectives

The goals, assumptions, and objectives of PIE and Safety Testing activities are as follows.

3.3.1.1 Goals

- Collect relevant fuel PIE and Safety Testing data, with the required accuracy, as a function of temperature, burnup, fast fluence, and coolant chemistry for developing fuel performance and fission product transport models, and to demonstrate acceptable fuel behavior under normal operating and potential accident conditions.
- Cooperate with other DOE-NE programs, and use international collaboration as much as possible to resolve key design data needs and minimize duplication of effort.

3.3.1.2 Assumptions

- HTGRs will be designed such that the radionuclides are substantially retained within the coated fuel particles during normal operation and all design basis accidents.
- Water or moisture ingress accidents are mitigated to have only moderate ingress flow rates rather than core flooding.
- Air and moisture ingress accidents are to be considered.
- DOE-NE will implement the requisite cooperative agreements to facilitate cooperation with other DOE-NE programs as well as international cooperation.

3.3.1.3 Objectives

- Quantify fission product release from the fuel during normal operation to the extent possible by analyzing the capsule components, fuel compacts, and fuel particles.
- Perform post-irradiation safety tests of fuel compacts in helium and oxidizing atmosphere to quantify fission product release; perform post-test destructive analysis of the compacts to assess fission product release and the effect of test conditions on particle integrity.
- Improve understanding of TRISO fuel behavior based on observed and measured phenomena that affect fuel performance and fission product release.
- Collect data to allow assessment of the design methods used to predict fuel performance to prescribed accuracy limits in a manner acceptable to regulators and stakeholders.

3.3.2 PIE Capabilities Development

In most cases, the major PIE and Safety Testing design data needs are sufficiently well known and lead directly to the measurements or tests to be performed to satisfy them. In some cases, development of a new measurement technique is required to satisfy a specific design data need, which leads to a task to develop or apply that new technique. In general, the PIE involves disassembly of test trains and capsules, nondestructive examination of fuel compacts and selected capsule components, measurement of fission product inventory on capsule components, destructive examination of fuel compacts and particles, and high-temperature safety testing of fuel compacts. Detailed discussion of specific activities for each irradiation experiment are given in Section 3.3.4.

Prior to initiating the AGR-1 PIE, the capabilities of candidate facilities at INL and ORNL were assessed. HTGR fuel has been examined and tested at ORNL since the 1960s. The ORNL hot cells and Core Conduction Cooldown Test Facility (CCCTF) have a full range of capabilities to support the required examinations and safety testing. INL hot cells have also been used to examine a wide variety of irradiated fuels for many years, including TRISO-coated lithium-target particles for tritium production in the NPR program. The relevant facilities at INL and ORNL were operating and functional at the beginning of this program, and both laboratories had development staff capable of designing, procuring, and installing the equipment, as well as developing the protocols for new or additional examination methods required for the AGR Fuel Program.

In some cases, new equipment has been developed and deployed to meet programmatic data needs. In addition, existing facilities and capabilities have been upgraded where necessary. This required preparation of new procedures for operations personnel, as well as the necessary environmental, safety, and health documentation, which was prepared to protect workers, the public, and the environment.

Equipment and enhancements were added to INL capabilities, including the following:

- Test train and component disassembly tools (in most cases, these are unique to a specific irradiation experiment).
- Remotely operable metrology equipment for each experiment.
- A fuel accident condition simulator (FACS) furnace in the Hot Fuel Examination Facility (HFEF) hot cell, including new feed-throughs and cabling to the FACS furnace for remote operation and data collection, and the associated Fission Gas Monitoring System located in the HFEF north corridor.
- Deconsolidation and LBL hardware and a dedicated remote particle handling microscope in the Analytical Laboratory hot cells.

- Equipment for performing stepwise, radial deconsolidation of AGR-3/4 fuel compacts in the Analytical Laboratory hot cells (necessary in order to remove driver particles and matrix from the outer regions of the compact while leaving the DTF particles untouched).
- Modifications to the HFEF precision gamma scanner, including installation of a Compton shield.
- Out-of-cell gamma counting equipment on the main floor of HFEF (used to gamma count FACS condensation plates and irradiated fuel particles) to increase throughput and decrease analysis time.
- Replacement of existing camera equipment in the HFEF main cell with a digital camera.
- Electron probe micro-analyzer for use in advanced microscopy in the Irradiated Materials Characterization Laboratory at MFC.
- Advanced focused-ion beam instrument with scanning electron microscopy capabilities for installation in the Irradiated Materials Characterization Laboratory.
- Capability for re-irradiating loose particles and whole compacts in the HFEF NRAD reactor to generate short-lived fission products (e.g., I-131 and Xe-133) in the fuel prior to safety testing.

Equipment and enhancements were also added to ORNL capabilities, including the following:

- Second-generation advanced irradiated microsphere gamma analyzer for gamma counting of individual particles.
- Upgrades to the particle micro-manipulator in the hot cell cubicle that also houses the irradiated microsphere gamma analyzer.
- Addition of an interlock system to the CCCTF furnace system, allowing hot exchange of the cold finger deposition plates for time-dependent data collection for fission product release.
- An upgrade to the CCCTF sweep gas-monitoring system.
- An upgrade to the gamma counting hardware.
- Improved liquid nitrogen supply system for the CCCTF fission gas system.
- Additional scanning electron microscopy capability, including a new energy dispersive x-ray spectroscopy detector.
- Deconsolidation/LBL system installed in the Irradiated Fuels Examination Laboratory.
- Addition of a Struers MiniMet polishing system to materialography capability.
- Addition of a customized shielded sample enclosure to the x-ray tomography system.

Procedures and instructions were developed for, and personnel were trained on, equipment and processes to meet NQA-1 requirements at INL and ORNL prior to their use.

In addition, a new furnace system for conducting post-irradiation heating tests in oxidizing atmospheres while measuring fission product release is being developed at INL. The Air/Moisture Ingress Experiment (AMIX) system will be used to evaluate fuel particle performance and fission product transport in atmospheres containing air or moisture at temperatures as high as 1650°C. The data represent a critical need for the program, making a safety case for the fuel under accident conditions that involve air or moisture ingress to the reactor core. The furnace system will be deployed in the Fuel Conditioning Facility (FCF) air cell.

3.3.3 PIE and Safety Testing Scope of Activities

The tasks associated with PIE and safety testing are discussed below. As noted earlier, some PIE tasks may not be required depending on results as the activity proceeds, but costs are based on completed and currently planned PIE and safety testing to provide the best estimate for program planning purposes. Determining the required tasks for a particular test train occurs during preparation of the PIE and safety test plan. Adjustments to the plan are made throughout the PIE and safety testing campaign based on results obtained during earlier examinations and testing and on budgetary considerations. Whether a full range of examinations is required for fuel irradiated under the AGR Fuel Program depends on many factors, including the defective fuel fraction measured during manufacturing, the in-pile R/B measurements, and the rate of coating layer failure observed during post-irradiation safety testing. If the fuel manufacturing effort is successful, the fuel being tested should have few, if any, defective particles (a fraction of exposed uranium $<10^{-4}$) and a low in-pile R/B ($<10^{-6}$). PIE will primarily address metallic fission product release fractions, distributions within the fuel and graphite, and coating layer behavior. It will also utilize the available capabilities to locate and examine failed fuel coatings within particles. Similarly, the level of post-safety-test analysis is somewhat dependent on the extent of coating layer failure during the tests, with extra effort expended to locate and understand the causes of failures. Cost estimates and tentative schedules for conducting PIE and Safety Testing are provided in Section 4.

Generally, the nominal time to complete PIE and Safety Testing of an irradiated test train is about four years, assuming that facilities and personnel are available. AGR-1 PIE and Safety Testing have taken place over about five years because of various issues that arose during the initial setup and performance of the multiple activities and the learning curves associated with them, as well as the expanded scope of the AGR-1 PIE (see Section 3.3.5). The overall AGR Fuel Program irradiation schedule has resulted in AGR-2 and AGR-3/4 PIE and safety testing commencing within about nine months of each other. The sharing of PIE and safety test work at the INL and ORNL sites is necessary to handle the workload. This is most pressing for complex, time-consuming tasks, such as the safety tests, which involve high-temperature heat-up for extended periods followed by detailed fuel examination. The AGR-2 PIE and Safety Testing are being performed primarily at ORNL, while AGR-3/4 PIE is being performed primarily at INL. These work activities are split because of additional capabilities at ORNL for destructive examination and particle analysis; a lack of AGR-3/4 sample transport methods from INL to ORNL for the graphite, matrix rings, and fuel bodies; and time constraints to complete PIE on both experiments.

The tasks described in this section summarize all of the basic PIE and safety testing activities expected to be performed as part of the AGR program.[b] The planned tasks expected to be performed for each particular irradiation experiment are shown in Table 2. The primary goal is to ensure that the needed measurements and tests are accomplished with the required level of accuracy. If this is impractical, the program needs early notification so alternative actions can be taken. In particular, some data may prove to be very expensive or time-consuming to collect, and different approaches to modeling or fuel qualification may have to be explored.

b. Previous versions of this Plan included an activity titled “Properties of irradiated materials specimens,” which involved collecting thermal, physical, and mechanical properties on various materials. Due to the complexity of performing characterization on materials contaminated with fission products, this has been pursued to a limited extent by inserting non-fueled specimens (e.g., pure compact matrix specimens) in Advanced Graphite Creep (AGC) Program irradiation experiments. It has, therefore, been removed from this plan, since no such measurements are planned on materials from AGR irradiation experiments.

- *PIE TASK-1: Test train receipt and visual inspection.* The transfer and nuclear accountability documentation will be completed, and the HFEF truck lock will be prepared for the receipt of the GE-2000 cask or other approved shipping configuration containing the test train. The shipment will be transported from ATR to the HFEF truck lock. The test train will be removed from the cask in the truck lock and moved into the HFEF hot cell where photo-visual examination of the test train will be conducted. The AGR-1 and AGR-2 test trains have been shipped in the GE-2000 cask as individual shipments. The length of the AGR-3/4 test train and the internal dimensions of the GE-2000 cask required that it be shipped in two sections and two shipments. It is anticipated that this shipping configuration will also be required for the AGR-5/6/7 test train if the GE-2000 cask or a similar shipping system is used.
- *PIE TASK-2: Test train nondestructive examination*. The intact test train will be analyzed in the HFEF main cell using the precision gamma scanner for a high-resolution gamma scan in the axial direction to help verify the position of the test train's internal components. Neutron radiography in NRAD may also be used to perform nondestructive examination of the test train's internal components.
- *PIE TASK-3: Test train and capsule disassembly*. The test train and capsules will be disassembled in the HFEF hot cell using in-cell disassembly equipment, tools, and jigs to remove the fuel compacts and internal components and store them in containers until needed for subsequent PIE activities.
- *PIE TASK-4: Component metrology*. The fuel compacts and internal capsule components will be visually and dimensionally inspected in the HFEF hot cell. After completion of this task for each of the AGR-1,[48] AGR-2,[49] and AGR-3/4[25] experiments, a "first look report" will be issued with extensive photographs and descriptions of the initial findings regarding the physical appearance of the test trains and components and the results of dimensional measurements.
- *PIE TASK-5: Compact shipments to ORNL.* Selected compacts of interest will be packaged and shipped from INL to ORNL for concurrent PIE and Safety Testing. Shipments of compacts to ORNL are made in approved shipping packages by a commercial carrier. Twenty AGR-1 compacts have been shipped to ORNL for PIE and Safety Testing. Twenty AGR-2 compacts have been shipped from INL to ORNL. Six AGR-3/4 compacts have been shipped from INL to ORNL for PIE work, with plans to ship four additional compacts in the next fiscal year. AGR-5/6/7 compact shipping plans will likely be similar to AGR-2 depending on available approved shipping packages.
- *PIE TASK-6: Graphite fuel holder and graphite/matrix ring gamma scanning.* Empty graphite fuel holders from AGR-1 and AGR-2 have been gamma scanned to quantify total inventory and identify potential hot spots from fission product release. AGR-3/4 graphite and matrix rings have also been gamma scanned to determine the inventory and distribution of fission products retained in the rings. AGR-5/6/7 graphite holders will be scanned in a similar manner for fission products to quantify total inventory and identify any locations with elevated activity that may be indicative of compacts containing particles with failed SiC. If detected, the fission product distribution will be mapped to determine the location of hot spots.
- *PIE TASK-7: Fuel compact gamma scanning.* Fuel compacts will be characterized with gamma spectroscopy to determine inventories of key fission products and measure fuel burnup.
- *PIE TASK-8: Melt and flux wire analysis.* Melt and flux wires will be removed from the graphite holders and analyzed to determine neutron flux levels and possible indications of high temperatures. The wires will be analyzed at either Pacific Northwest National Laboratory (the original manufacturer

of the melt and flux wire packages) or INL. Note that AGR-5/6/7 contains no internal melt or flux wires.

- *PIE TASK-9: Capsule deposited fission products.* Irradiation capsule components will be analyzed for fission products in order to determine the total release from the fuel compacts. This includes analysis of fission product in/on: the interior metal surfaces of each capsule shell and the through-tubes (except in the case of AGR-3/4, where the cold graphite sink ring is expected to act as a barrier to fission product migration to the steel capsule shell); the graphite fuel holders; and additional ancillary components such as graphite and grafoil spacers.
- *PIE TASK-10: Radionuclide transport in irradiated specimens.* Radionuclide inventory and distribution in irradiated AGR-3/4 matrix material and graphite specimens will be measured using appropriately established techniques, such as beta and gamma spectrometry and physical sampling and analysis.
- *PIE TASK 11: Microanalyses of fuel compacts.* Selected compacts from irradiation (after safety testing) will be analyzed in cross sections at the microscopic scale to assess localized effects of irradiation and post-irradiation heating on the compact matrix and embedded fuel particles. This has been completed for AGR-1 compacts, is under way for AGR-2 and AGR-3/4 compacts at the time of this writing, and will be performed on AGR-5/6/7 compacts.
- *PIE TASK-12: Compact deconsolidation.* Selected compacts from each of the experiments will be deconsolidated to free individual fuel particles from the matrix binder as a precursor to the LBL process and to provide loose fuel particles for other PIE tasks.
- *PIE TASK-13: Compact LBL.* The standard procedure is to perform an initial acid leach on deconsolidated compacts, particles, and matrix to dissolve uranium and fission products in the matrix and exposed kernels. The particles and matrix debris are exposed to air at elevated temperatures (750°C) to oxidize matrix and PyC material not protected by intact SiC coatings. A post-burn leach will then be performed to dissolve any additional fission products that were present in the matrix debris or the OPyC layer and to dissolve uranium and fission products exposed by the burn step. These tasks will usually be combined with *PIE TASK-12* above.
- *PIE TASK-14: Particle inspection and sorting.* Intact particles from deconsolidation and/or LBL will be optically examined with sufficient magnification to provide an indication of the surface condition of the particles.
- *PIE TASK-15: Burnup measurement.* The primary means of burnup measurement is activity ratios determined from the compact gamma scans in *PIE TASK-7*. Destructive isotopic analysis methods will be used on particles from selected compacts as a benchmark to compare with the burnup determinations from the gamma scanning data.
- *PIE TASK-16: Irradiated microsphere gamma analysis.* Individual particles from each of the experiments will be gamma counted to quantify the inventories of selected gamma-emitting fission products. The data will be used to gauge the relative fission product retention in each of the analyzed particles and can be used to screen for failed particles based on radionuclide inventories before performing other analyses.
- *PIE TASK-17: Microanalysis of fuel particles.* Particles identified in the previous tasks will be prepared in cross section for individual examination, including optical microscopy, scanning electron microscopy, transmission electron microscopy, high-resolution transmission electron microscopy, scanning transmission electron microscopy, and chemical analysis using EDS and wavelength

dispersive spectroscopy. This task may also include analysis of intact particles using x-ray tomographic methods. The objective of this task is to examine kernel and coating microstructures at a range of length scales and identify fission product and actinide distributions in the various layers to better understand particle behavior and fission product transport.

- *PIE TASK-18: Safety testing – re-irradiation.* Selected compacts or particle samples will be re-irradiated before safety testing, primarily to generate short-lived fission products, including ^{131}I and ^{133}Xe, so I and Xe release during safety testing can be measured. Reirradiation will take place in the NRAD reactor at MFC, and subsequent heating tests will utilize the FACS furnace.
- *PIE TASK-19: Safety testing.* Selected compacts will undergo heat-up tests in helium at peak temperatures of approximately 1400 to 1800°C for planned durations of approximately 300 consecutive hours. Both isothermal and variable temperature profiles will be used. Gaseous fission product release will be measured continuously during the tests, and condensable fission product release will be measured by analysis of condensate surfaces within the furnaces that are periodically replaced and analyzed for deposited isotopes. A separate fuel safety testing capability, AMIX, is being developed to extend the chemical environment capabilities to temperatures up to approximately 1650°C in an oxidizing atmosphere typical of air- and moisture-ingress events. The AMIX capability will be used to test AGR-5/6/7 fuel compacts. If the capability is established significantly before the availability of AGR-5/6/7 compacts, archived AGR-2 compacts can be used. Additional tests in oxidizing atmospheres will be performed using archived, irradiated AGR-3/4 fuel compacts, focused specifically on the effect of air and moisture on fission product release from exposed kernels. Additional discussion of the oxidation tests is provided in Section 3.3.6.
- *PIE TASK-20: Graphite and matrix heating tests.* Irradiated graphite and matrix specimens, with fission products deposited in them during irradiation, will be heated in a variety of atmospheres (potentially including dry helium and helium with various concentrations of air or moisture) while measuring fission product release. Tests in helium can also be used to help derive diffusion coefficients for various fission products from the rings.
- *PIE TASK-21: Archiving and waste handling*. Some fuel specimens in various configurations (kernels, TRISO particle fuel, and compacts) will be collected and placed into archives at INL and ORNL for further research or historical purposes. Residual materials not chosen for archival storage will be handled as waste. Collecting, packaging, and disposing of irradiated fuel specimens and associated waste generated during AGR PIE will take place at ORNL and INL. The type of waste involved will determine its need for treatment or its disposition path.
- *PIE TASK-22: Reporting*. Researchers will disseminate the findings, results, and lessons learned from the PIE task in formal and informal reports, presentations, and publications. Also, support will be provided for program requests for specific information, clarifications, and impact assessments.

3.3.4 Test-Train-Specific PIE and Safety Testing

An assessment of the applicability of the detailed PIE and Safety Testing tasks defined above for the individual irradiation test trains, based on the objectives of each test train, resulted in the task assignments shown in Table 2. The objectives of the PIE and Safety Testing of each test train are summarized in Subsections 3.3.5 through 3.3.5.3. Actual and estimated costs and schedules for PIE and safety testing of each test train are provided in Section 4.

3.3.5 AGR-1: Shakedown Test – PIE of Test Train and Early Fuel

As previously noted, the initially planned purpose of AGR-1, the first test train to undergo irradiation, PIE, and safety testing, was to gain experience with multi-monitored capsule test train design, fabrication, and operation, and to reduce the chances of test train or capsule failures in subsequent test trains. An additional purpose was to reestablish, develop, and shake down PIE and safety testing equipment and methods to be used for later experiment irradiations. However, the scope of AGR-1 PIE was substantially expanded to:

- Provide extensive data on fuel performance under irradiation and simulated accident testing to support specification of the fuel to be qualified in later irradiation test trains
- Support early HTGR pre-licensing interactions with the NRC
- Develop a quantitative understanding of the relationships among the fuel fabrication processes, fuel product properties, and irradiation performance.

The specific PIE and safety testing tasks performed on this test train are identified in Table 2.

3.3.5.1 AGR-2: PIE of Fuel Performance Test Train. The AGR-2 PIE and safety testing is providing irradiated fuel performance data beyond the online R/B measurements for UCO and UO_2 fuel types fabricated in the larger production-scale (6-in.) coater, as discussed in Subsection 3.2.3.2. The PIE and safety testing also support development of a fundamental understanding of the relationships among fuel fabrication processes, fuel product properties, and irradiation performance. The specific PIE tasks and safety test tasks performed so far or planned to be performed on this test train are identified in Table 2.

Table 2. Test train PIE tasks.

Task Number	Task	AGR-1	AGR-2	AGR-3/4	AGR-5/6/7
PIE TASK-1	Test train receipt and visual inspection	X	X	X	X
PIE TASK-2	Test train nondestructive examination	X	X	X	X
PIE TASK-3	Test train and capsule disassembly	X	X	X	X
PIE TASK-4	Component metrology	X	X	X	X
PIE TASK-5	Compact shipments to ORNL	X	X	X	X
PIE TASK-6	Graphite holder gamma scanning	X	X	X	X
PIE TASK-7	Fuel compact gamma scanning	X	X	X	X
PIE TASK-8	Melt and flux wire analysis	X	X	X	
PIE TASK-9	Capsule deposited fission products	X	X	X	X
PIE TASK-10	Radionuclide transport in irradiated specimens			X	
PIE TASK-11	Microanalysis of fuel compacts	X	X	X	X
PIE TASK-12	Compact deconsolidation	X	X	X	X
PIE TASK-13	Compact LBL	X	X	X	X

Table 2. (continued).

Task Number	Task	AGR-1	AGR-2	AGR-3/4	AGR-5/6/7
PIE TASK-14	Particle inspection and sorting	X	X	X	X
PIE TASK-15	Burnup measurement	X	X	X	X
PIE TASK-16	Irradiated microsphere gamma analysis	X	X	X	X
PIE TASK-17	Microanalysis of fuel particles	X	X		X
PIE TASK-18	Safety testing –re-irradiation		X	X	X
PIE TASK-19	Safety testing	X	X	X	X
PIE TASK-20	Graphite and matrix heating tests			X	
PIE TASK-21	Archiving and waste handling	X	X	X	X
PIE TASK-22	Reporting	X	X	X	X

3.3.5.2 AGR-3/4: PIE of Fission Product Transport Test Train. The AGR-3/4 PIE and Safety Testing will provide data to support the calculation of fission product diffusivities in fuel kernels and coated particles, as well as fission product diffusivities and sorptivities in fuel compact matrix and graphite for use in upgrading fission product transport models and codes. This PIE will focus on measurements of fission product inventories and concentration profiles in the graphitic components, including a full mass balance to support fission product transport model development. However, the PIE activities will also involve nondestructive and destructive examination of selected fuel compacts to (a) obtain an inventory and radial distribution of fission products in the compact matrix, (b) obtain and examine loose particles, and (c) examine DTF and driver fuel particles within cross sections of the compacts. In addition, heating fuel compacts and irradiated graphite and matrix materials will be performed in a variety of test atmospheres (including dry helium, air, and moisture) while measuring fission product release. The specific PIE and safety test tasks performed so far, or planned to be performed, on this test train are identified in Table 2.

3.3.5.3 AGR-5/6/7: PIE of Fuel Qualification and Fuel Performance Limits Test Train. This is now planned to include three experiments (i.e., AGR-5/6/7) in a single test train. The AGR-5/6/7 PIE and Safety Testing will document fuel integrity and safety test performance to demonstrate compliance with statistical performance requirements under normal operating and potential accident conditions. The primary interest is to verify successful fuel performance. This PIE makes heavy use of the fuel heat-up capabilities. The AGR-7 PIE measures the capability of the selected fuel to withstand irradiation and potential accident conditions beyond the conditions in AGR-5/6 in support of plant design and licensing. The specific PIE and safety test tasks planned for this test train are identified in Table 2.

3.3.6 Testing to Assess the Impacts of Air and Moisture Ingress on Fuel Material

Air and moisture ingress into the HTGR core can occur during various accident scenarios. Air ingress may occur following a depressurization accident. The severity of the event depends on break size, break location, and design of the reactor cavity, all of which influence the ability of air to enter the core via natural circulation, stratified flow, or molecular diffusion. Reactor designs that include a steam generator in the primary coolant loop introduce the risk of steam generator tube leaks, resulting in moisture ingress into the primary coolant system. The effects of these oxidants on the fuel behavior are important considerations. Among the effects of importance for understanding and predicting fission product release from the core during these events are oxidation rates of the matrix, TRISO particle integrity during high-temperature exposure to oxidants, hydrolysis of exposed kernels and the subsequent release of fission products, and volatilization of fission products from fuel matrix and core graphite.

The AGR Fuel Program plans to conduct experiments to collect important data on these effects. This will include both in-cell tests performed on irradiated fuels and related materials and out-of-cell tests performed on unirradiated materials. A research plan for assessing the effects of air and moisture ingress was prepared previously [35] and will be used as a guide for developing a specific test plan as part of the AGR Fuel Program. The overall goals for these tests are to establish the effects of oxidants on fuel behavior and provide data that can be used to better predict fuel performance and fission product transport within the core.

A dedicated in-cell furnace is being developed at INL for testing irradiated specimens.[50] The AMIX system will be capable of heating specimens to temperatures in excess of 1600°C in a variety of oxidizing atmospheres, including both air and moisture over a large range of partial pressures, as well as pure helium. The system will allow online measurement of fission gas released from the specimens using dedicated traps and gamma detectors. It will also include the capability to collect and measure condensable fission products. The system will be installed in the FCF air hot cell. Specimens intended for use in this system include the AGR-2, AGR-3/4, and AGR-5/6/7 fuel compacts, the AGR-3/4 matrix and graphite rings, and the AGR-3/4 fuel bodies. Additional specimens may be identified as planning continues. Current estimates are that the AMIX system will begin tests with irradiated fuel at the end of 2021 or the beginning of 2022.

Testing is also being performed using unirradiated matrix specimens heated in steam atmospheres. Additional research on air and moisture oxidation of SiC and matrix continues via several university proposals that are currently in their second of three years.

3.4 Fuel Performance Modeling

A key product of the AGR Fuel Program is the development of fuel performance models. As discussed here, fuel performance modeling addresses the structural, thermal, and chemical processes that can lead to coated particle failures. The modeling considers the effects of fission product chemical interactions with the coatings, which can lead to degradation of the coated particle properties. Fission product release from the particles and transport within the fuel-compact matrix and fuel-element graphite are also modeled. Many groups have attempted to model the performance of coated particle fuels.[51] These efforts have resulted in empirically driven models that are limited in application to environmental conditions, fuel forms, and design configurations closely resembling those the empirical data are based around. The most significant reasons the modeling has been limited in application are (1) incomplete representative coating property data as a function of irradiation conditions and (2) insufficient understanding of the interactions between phenomena as irradiation proceeds. Thus, the goals are to:

- Develop fuel performance models of coated particle fuel (either UCO or UO_2) that are more first-principle based and can be used to:
  - Guide current and future particle designs

  - Assist in irradiation and safety test experiment planning
  - Predict observed fuel failures and fission product release
  - Allow more accurate interpolation of fuel performance inside the performance envelope needed for core design assessments and modest extrapolation of fuel performance outside the existing performance envelope when required

- Develop a prioritized list of material properties and constitutive relations needed for accurate modeling of coated particle fuel under normal and off-normal conditions
- Develop advanced models that take advantage of new methods
- Benchmark these models/codes against U.S. and international irradiation and safety test experiments, where possible.

The effort by the modeling working group has been focused on improving these crucial areas. Performance modeling is an iterative task. Work began on modeling during the days of the Dragon Project in the 1960s and continued through the 1990s, as documented in the results of an International Atomic Energy Agency (IAEA) Coordinated Research Project on fuel performance and fission product behavior.[52] More recently, another IAEA Coordinated Research Project code-to-code benchmark was conducted with improved models.[52] While useful, currently available models are not adequate for the applications mentioned earlier. Models will continue to evolve throughout the fuel development phase and into the period of commercial fuel manufacturing and power generation. This has been the case with every reactor system deployed for electricity production.

Fuel performance models are used for: (1) assisting in developing candidate coated particle fuel designs, (2) predicting the performance of coated particle fuel during irradiation testing and post-irradiation heat-up, and (3) calculating fuel performance for HTGR core designs under normal operating and hypothetical accident conditions. Developing fuel performance models requires fundamental understanding of potential failure mechanisms and how these mechanisms depend on the irradiation conditions and the material constituting the fuel. Accurate fuel performance modeling will also require good material properties and constitutive relations information.

Table 3 summarizes the key fuel failure mechanisms associated with TRISO particle fuel and how these mechanisms depend on reactor service conditions, particle design, and performance parameters. The failure mechanisms considered under irradiation are (1) pressure vessel failure; (2) cracking of the IPyC layer and IPyC layer partial debonding, leading to cracking of the SiC layer; (3) kernel migration; and (4) diffusive release through intact layers. Under hypothetical accident conditions, the failure mechanisms considered are (1) fission product attack of the SiC; (2) SiC thermal decomposition; (3) increase in SiC permeability/SiC degradation; (4) oxidation of the SiC layer; and (5) rapid energy deposition.

Table 4 summarizes the important material properties required for accurate modeling under irradiation and potential accident conditions and lists the state of knowledge of the specific properties, their importance to modeling, and potential measurement techniques. The ability to obtain measurements for all of these material properties is limited by program resources and, in some cases, by measurement science, given the size of the TRISO particle, its individual constituents, and the nature of the actual measurement to be made.

The scope of this section is limited to activities needed to support fuel performance modeling. However, as indicated in Table 3, fission product release from the kernel and transport of fission products through the coating layers directly affect some failure mechanisms. The source term aspects of fission product transport behavior are covered under the Fission Product Transport and Source Term element of the program.

The R&D needs for fuel performance modeling are briefly summarized in subsections 3.4.1–3.4.6. The activities required to address these needs (fabrication of test articles, irradiation, and PIE) are addressed in the appropriate program element, with more detailed planning performed as the program proceeds.

Table 3. Summary of coated particle failure mechanisms.

Failure Mechanism	Reactor Service Conditions	Particle Design and Performance Parameters	Comments
Pressure vessel failure	Temperature Burnup Fast fluence	Strength of SiC Buffer density (void volume) Fission gas release CO production Particle asphericity Layer thicknesses Kernel type (UO_2, UCO)	
Irradiation-induced PyC failure	Fast fluence Temperature	Dimensional change of PyC Irradiation-induced creep of PyC Anisotropy of PyC Strength of PyC PyC thickness PyC density	
IPyC partial debonding	Temperature Fast fluence	Nature of the interface Interfacial strength Dimensional change of PyC Irradiation-induced creep of PyC	
Kernel migration	Temperature Burnup Temperature gradient	Layer thicknesses CO production Kernel type (UO_2 versus UCO)	Modeled with semi-empirical measured migration coefficient.
Diffusive release through intact layers	Temperature Burnup Temperature gradient Time at temperature	Chemical state/transport behavior of fission products Microstructure of SiC SiC thickness	Could be more important at high burnup in LEU fuels because of greater yields of Pd from Pu fissions and because of higher temperatures in future designs. More important under accident conditions. AGR-1 safety testing results indicated increased release of Ag and Eu from intact particles with the fine-grained SiC microstructure variant after ~100 h at 1800°C.

Table 3. (continued).

Failure Mechanism	Reactor Service Conditions	Particle Design and Performance Parameters	Comments
Corrosion of SiC by CO	Temperature Burnup Time at temperature	Kernel type (UO_2, UCO) IPyC performance	CO is generated in particles with UO_2 kernels. At elevated temperatures, CO can attack the SiC layer if the IPyC layer is porous or has failed.
SiC thermal decomposition	Temperature Time at temperature	SiC thickness Microstructure of SiC	Not important in traditional accident envelope (peak temperature <1600°C). Expected to be important at ~2000°C. Degradation observed at 1800°C in coated particles was attributed to this mechanism but may be attributed to fission product attack instead.
Increase in SiC permeability/SiC degradation	Burnup Temperature Fluence	Microstructure of SiC[a] Diffusion[a] Buffer densification and cracking[a] Thickness of SiC Permeability of SiC	Exact mechanism is unclear, but limited data from higher burnup fuel suggest increased fission product release under long-term heat-up. Could be fission product attack and would be more important at higher burnup in LEU fuels because of greater yields of Pd from Pu fission and higher operating and/or accident temperatures.
Oxidation of SiC layer	Partial pressure of oxygen Temperature Time at temperature	Thickness of SiC layer Microstructure of SiC layer	Results from external attack such as air or water. Needed for modeling kinetics of oxidation.
Rapid reactivity insertion	Energy deposition (J/g-fuel) Time duration of the deposition Burnup of fuel	Degree of kernel melting/vaporization Thickness of layers Coefficient of thermal expansion of layers Elastic modulus of layers Swelling of kernel Kernel-coating mechanical interaction	Limited data available. However, available data indicate that reactivity events in an HTGR are relatively benign in comparison to other technologies.

a. Indicates a parameter with uncertain effect on the failure mechanism

Table 4. Key material properties needed for fuel performance modeling.

Property	Current State of Knowledge	Importance in Modeling	How to Measure
Irradiation performance			
PyC anisotropy	Known to be critical to characterize PyC behavior. Ability to measure it accurately and precisely is needed.	All key IPyC properties are thought to depend on the anisotropy.	Use x-ray, Raman laser, and optical methods.
PyC irradiation-induced dimensional change	Reasonably well known as a function of temperature and density. Key issue is link between shrinkage and anisotropy.	Stress depends on ratio of shrinkage rate to irradiation-induced creep.	Measure dimensional change on PyC specimens.
PyC irradiation-induced creep	Uncertain with a factor of 5, based on limited database. Would like to know creep as a function of temperature, density, and anisotropy.	Stress depends on ratio of shrinkage rate to irradiation-induced creep.	Use special specimens (split composite ring test).
Poisson's ratio in creep	Reasonably well known. Literature data range from 0.3 to 0.5. Best estimate is 0.4. Probably a function of density. Unclear whether it is a function of anisotropy.	Has modest effect on stress in PyC layer.	Use special specimens.
Strength of PyC	Data vary significantly. Some exist as a function of density and anisotropy. Key issue is how well the anisotropy of the PyC was known, because that determines the functional relationship.	Very important.	Obtain bistructural-isotropic-coated particles that can be tested using classic ring test or crush test.
Strength of SiC	Data vary significantly. Need data as a function of density, neutron fluence, irradiation temperature, and microstructure (large grain versus small grain and columnar versus equiaxial). Microstructure is a function of deposition conditions. Data are available for Chinese SiC. German data suggest that irradiation can reduce strength. The U.S. has correlated many data and concludes there is still uncertainty about effect of irradiation. There are non-trivial issues related to experimental procedures used in past measurements. The presence of free Si in the SiC layer can cause strength reductions.	Very important.	Can use irradiated particles as well as classic brittle ring technique. Also can use axial compression of a cylindrical plug inside SiC cylindrical sample. Key issue is linkage of data to microstructure.

Table 4. (continued).

Property	Current State of Knowledge	Importance in Modeling	How to Measure
Interfacial bond strength between SiC and PyC	Very little is known. Historic value of ~50 MPa is used in calculations. Recent data that simulated SiC/PyC bond indicated strengths of 50 to 100 MPa. Tends to agree reasonably well with values from SiC/SiC composites.	Needed to understand the nature of debonding of the layers. The nature of the bond depends on the nature of the fabrication process.	Use special specimens and special punch/shear test to get bond strength.
Irradiation-induced swelling of SiC	Data are being obtained in U.S. fusion program. Swelling is on the order of 0.2 to 1.2% in temperature range of interest.	Lower importance given uncertainty in other parameters.	Take density (density gradient column) measurements.
Irradiation-induced SiC creep	Limited data at low fluence.	Modest impact. PyC creep is much larger effect.	Use split-ring or bend-strength relaxation techniques.
Fission gas release from the kernel	Data on gas release are reasonably well known for UO_2. Little to no data on UCO, especially at high burnup.	Direct contributor to pressure in particle.	Can be measured by crushing particles or online from "intentionally failed" particles.
CO production	Important for UO_2 fuel only. Data exist at low burnup from German program. No data at high burnup.	Direct contributor to pressure in particle and affects kernel migration.	Can be measured by crushing particles.
Kernel swelling	Reasonably well known at moderate burnup. More data at very high burnups would be useful.	Need to prevent kernel/ coating mechanical interaction.	Part of PIE planning for irradiated fuel.
Accident performance: long-term heating/air ingress/rapid reactivity transients			
Thermal expansion coefficient of PyC	Thermal expansion is different in the two orientations in PyC and depends on the anisotropy of the material. Effect of irradiation is not well known. Limited data available.	Critical for potential reactivity events where large temperature gradients may develop within the fuel particle.	Use conventional techniques. Small sample size adds to overall difficulty in measurement and uncertainty.

Table 4. (continued).

Property	Current State of Knowledge	Importance in Modeling	How to Measure
Elastic modulus of PyC	Modulus is a function of anisotropy, fluence, density, and temperature. Few to no data at very high temperatures expected in accidents.	Critical for potential reactivity events where large temperature gradients may develop within the fuel particle.	Use resonant ultrasound spectroscopy or nano-indentation.
Elastic modulus of SiC	Data from fusion program show a 10% drop at reactor-relevant temperatures and radiation doses. Little data above 1000°C.	Critical for potential reactivity events where large temperature gradients may develop within the fuel particle.	Use resonant ultrasound spectroscopy or nano-indentation.
Thermal expansion coefficient of SiC	Limited amount of data suggests expansion is constant between 900 and 1300°C. No systematic dependence on coating temperature or neutron irradiation. The presence of free carbon in SiC can reduce coefficient of thermal expansion by 40%.	Critical for potential reactivity events where large temperature gradients may develop within the fuel particle.	Use conventional techniques. Small sample size adds to overall difficulty in measurement and uncertainty.
Fission product interactions with layers and potential degradation of properties	Unknown influence at present.	Unknown at present.	Need to examine irradiated high-burnup particles that have been heated to determine the magnitude of the effect.
Buffer survivability	This effect needs to be studied with the performance model before a definitive direction on the need for this work can be determined.	Have some properties on buffer strength and dimensional change to determine its failure; these can be used as a starting point for evaluations.	Need to produce some low-density material for material tests.
Buffer-IPyC bond strength	No data are available. The current model presumes complete, early debonding between the buffer and IPyC layer. However, AGR data indicate that the buffer can remain adhered to the IPyC late into the irradiation in some cases.	Would be necessary to accurately model the observed SiC failure mechanism in which the buffer precipitates IPyC failure, which leaves the SiC layer vulnerable to focused fission product attack.	Yet to be determined.

Table 4. (continued).

Property	Current State of Knowledge	Importance in Modeling	How to Measure
Kernel swelling under rapid energy deposition	Little data available under rapid energy deposition conditions for reactivity-induced accidents that are more severe than anticipated for HTGRs. Some data has been collected by Japan under the GIF HTGR fuel collaboration.	Kernel swelling and kernel coating mechanical interaction may be critical to predicting failure in rapid reactivity transients. Low priority in the AGR program due to the low probability of rapid reactivity insertion in gas reactor designs.	Part of PIE following reactivity transient testing.

3.4.1 Thermomechanical and Thermophysical Properties of Coating Layers under Normal Operation

The thermomechanical and thermophysical properties of PyC and SiC listed in Table 4 are needed as a function of fast fluence and deposition conditions, where appropriate. In many cases, these measurements need to be made on samples of the individual materials because of the difficulty of making the measurement on the coated particle in situ. Examples of the properties include anisotropy of PyC, irradiation-induced dimensional change of PyC, irradiation-induced creep of PyC, PyC Poisson's ratio in creep, interfacial bond strength between SiC and PyC, irradiation-induced swelling of SiC, irradiation-induced creep of SiC, and Weibull strength of PyC and SiC. This work was initiated at ORNL and the University of Michigan but was halted in 2013 because of a lack of funding. No further work is planned at this time within the AGR Fuel Program. A European irradiation experiment was performed to explore pyrocarbon creep,[53] but no data have been published in the open literature.

3.4.2 Thermochemical Properties of Kernel under Normal Operation

The thermochemical properties of the kernel listed in Table 4 are needed as a function of burnup. Fission gas release from UO_2 kernels is reasonably well understood. Data on fission gas release from UCO kernels is needed over the relevant burnup and temperature ranges for the HTGR. Information on CO release from UO_2 kernels is also needed at burnups in excess of 10% FIMA at relevant reactor temperatures (up to 1300°C). Finally, measurements of kernel swelling for both UO_2 and UCO kernels are needed, especially at high burnup. Kernel swelling measurements have been performed on AGR-1 UCO kernels, and additional measurements are being performed on AGR-2 UCO and UO_2 kernels.

3.4.3 Thermomechanical and Thermophysical Properties of Coating Layers under Accident Conditions

Table 4 lists the properties needed to model the mechanical behavior of the coated particle under accident conditions. The thermal expansion coefficient and elastic modulus of PyC are needed as functions of fast fluence and temperature (1200 to 1800°C). The corresponding properties of SiC are also needed. Work in these areas is not planned under the existing budget scenario. No proposed locations or personnel have been identified to perform this work, should its priority increase.

3.4.4 Thermochemical Properties of the Kernel and Coating Layers under Accident Conditions

Fission products can interact with the SiC layer and degrade the properties of the layer. Of greatest concern is Pd attack under accident conditions. Many researchers have studied the attack of the SiC layer by Pd. The impact of the attack on the degradation of the properties of the layer has not been studied. Simple one-dimensional models assume that the particle fails when ~50% of the SiC layer has been attacked. A more sophisticated finite-element approach that models degradation and assesses the resulting thermomechanical response of the degraded coatings has been developed and is being implemented in the PARFUME code. Review of the historical data suggests that out-of-pile testing on ideal systems provides interaction rates that are orders of magnitude above that observed in coated particles.

Observations of fission product attack have been made during safety testing and PIE of AGR-1 and AGR-2 fuel compacts. No significant widespread attack of the SiC has been observed; rather, any notable fission product attack of the SiC layer (such that the microstructure is significantly degraded) takes place in a very localized manner, and only in cases where IPyC failure has occurred. Furthermore, it is not clear if SiC failure in these cases results in complete loss of retention of the layer (i.e., the layer becomes permeable to fission gases), or if it involves the creation of an interconnected carbon-rich pathway, which allows for rapid diffusive release of Cs (and potentially other condensable fission products) once it penetrates the entire layer. Because of the nature of the fission product attack (in particular the complex, thermomechanically driven failure of the IPyC layer that is required to initiate attack of the SiC layer), quantitative measurements of reaction thicknesses in the SiC layer have not been practical.

Data from Germany suggest that the SiC layer becomes permeable to certain fission products under high-temperature heating when the coated particles are exposed to higher-burnup and fast-fluence conditions (14% FIMA, 6 to 8×10^{25} neutrons/m^2). The permeability may be associated with a microstructural change or corrosion of the SiC by CO above a critical concentration. Alternately, the permeability may be a mischaracterization of the reason for the higher fission product releases because of uncertainties associated with the irradiation history (especially temperature) of the AVR pebbles that were tested. Further evaluation of the original data is needed.

Tests are planned to evaluate the oxidation behavior of SiC as part of the accident heat-up tests in AGR-5/6/7, in which the influence of air on fuel behavior will be studied. Low air partial pressures and fuel temperatures consistent with air-ingress calculations will be used. Additional studies on unirradiated particles are anticipated as part of a DOE funding opportunity announcement issued in FY 2018 (research to take place in the 2019 to 2021 timeframe).

Kernel swelling and kernel coating mechanical interaction may be critical for predicting failure in reactivity insertion accidents. These data can be obtained as part of PIE following reactivity insertion accident simulation testing. However, reactivity insertion accident testing is not currently planned as part of the AGR Fuel Program because the likelihood of rapid (super prompt critical) reactivity transients that could induce fuel failures are precluded by the current prismatic HTGR design.

3.4.5 Thermophysical and Physiochemical Properties of Fuel Compacts

With the AGR fuel compacting process for HTGR fuel, thermophysical and physiochemical properties of the compact need to be measured to enable accurate fuel performance assessments in the HTGR irradiations. Of these properties, the irradiation-induced shrinkage and the thermal conductivity of the compact as a function of fluence and temperature need to be measured during PIE. Irradiation-induced shrinkage of fuel compacts has been assessed for the AGR-1, AGR-2, and AGR-3/4 irradiation via relatively simple post-irradiation dimensional measurements. Some effort has been expended to develop methods for measuring bulk compact thermal conductivity,[54] but deployment of these methods in a hot cell for performing measurement on irradiated compacts is not being pursued due to cost and complexity. Several matrix-only specimens have been irradiated in the AGC series of experiments, and thermophysical properties (e.g., thermal conductivity and thermal expansion) may be performed on those in the future.

3.4.6 Code Benchmarking and Improvement

Currently, significant activity is taking place around the world to develop improved fuel performance codes under normal operating and potential accident conditions. The benchmarking of fuel performance codes took place under the auspices of the IAEA for both normal and potential accident conditions through 2008, based mainly on historical irradiations and safety tests.[52] Additional benchmarking is foreseen under the GIF/VHTR Fuel and Fuel Cycle Program Management Board based on the behavior of the current generation of TRISO fuel in current irradiations and on safety tests planned in the U.S. and other international programs. INL has completed pre-test predictions for the AGR-1[15], AGR-2,[55] and AGR-3/4[56] experiments. Safety test predictions have been completed for the AGR-1[15] and AGR-2[16] experiments; fission product transport parameter estimation for AGR-3/4 will be performed in conjunction with AGR-3/4 PIE. Pre-test predictions and post-test calculations will be performed for the

AGR-5/6/7 irradiation experiments. Similar sets of calculations will be performed for a subset of the safety tests using accident performance models, as determined by the AGR Fuel Program. The program will consider updates to fuel performance calculations as appropriate when the new material properties data in the earlier experiments become available. The performance test fuel, fuel qualification irradiations, and accident testing, along with planned material property irradiations (obtained via the DOE Nuclear Energy Research Initiative and international collaborations or by irradiation of material samples in HFIR at ORNL) and other DOE-funded university research, may provide some of the separate-effects data needed to improve the fuel performance models.

3.5 Fission Product Transport and Source Term

The goal of the fission product transport and source term activity was to produce a technical basis for radionuclide source terms under normal and potential accident conditions for the HTGR. Initial studies were performed to measure hydrogen and tritium permeation into various high-nickel superalloys that could potentially be used in an HTGR. Reports and papers were published[57],[58] that discussed the outcome of these studies. However, work was halted in late 2011 due to DOE-NE's decision to defer further NGNP project work scope until a public-private partnership was firmly established. DOE-NE's recent announcement of a funding award to X-energy, LLC to lead a team that will pursue development of an HTGR is the first public-private partnership established in the U.S., with ORNL and INL listed as additional team members. As this award goes forward, further research may be performed in these areas in collaboration with the reactor designers. Under the INL ART program office, further work scope regarding fission-product transport and source term has been cancelled for lack of a selected reactor design.

3.6 Other Activities

A few other activities in the AGR Fuel Program are accounted for separately in the cost estimate in Section 4 because they do not fit easily into any one of the individual experiments or they cut across the different WBS elements in the program. These include:

- Reports that document the results of the AGR Fuel Program at key times will be given to the HTGR engineering design and licensing organizations for developing topical reports or producing safety documentation for the proposed plant.
- Facilities at ORNL and INL have been upgraded, and more upgrades may be required in the future to accomplish irradiation and PIE activities. The experience to date has been that some of the infrastructure needed to carry out the AGR Fuel Program was in need of repair/upgrade or did not exist. These upgrades and new capabilities have enabled the program to obtain the data outlined in the plan status update and path forward.
- Upgrades to the Nuclear Data Management and Analysis System software used to qualify and store all of the data generated in the AGR Fuel Program that incorporate the latest versions of underlying software and interfaces with the internet are anticipated over the remaining life of the program.

4. PROGRAM SCHEDULE AND COST

A detailed activity-based schedule (lifecycle baseline) for the activities presented in this technical program plan for TRISO fuel has been developed and is used to guide and prioritize activities year by year. A higher-level summary of that schedule is shown in Figure 3. The critical path for the fuel qualification remains through the AGR-5/6/7 irradiation at this time and then shifts to PIE and Safety Testing once the irradiations are complete. Irradiation durations are determined by their location in ATR. AGR-1 (620 EFPDs) and AGR-2 (559 EFPDs) were longer irradiations because of the lower thermal flux in the respective large B irradiation positions. AGR-3/4 had a much shorter duration (369 EFPDs), because it was irradiated in the NEFT and was a fission product transport test rather than a fuel qualification test. The AGR-5/6/7 irradiation will be approximately 500 to 550 EFPDs, because it will also be irradiated in the NEFT, a higher flux position in ATR. The AGR-5/6/7 irradiation is also a qualification and margin test for the final AGR-5/6/7 fuel. The durations for PIE and safety testing are based on: (1) estimates of throughputs at ORNL and INL based on the scope of anticipated activities, considering historical and current experience at INL and ORNL for AGR-1 and AGR-2 PIE and safety testing; (2) anticipated learning-curve effects for the safety testing and PIE of later compacts; and (3) schedule overlaps in the safety testing and PIE-related activities for fuel from each of these compacts, with consideration of PIE and safety testing experience gained with the early test trains. Based on the project schedule shown in Figure 3, the fuel for the HTGR is anticipated to be qualified by FY 2024, assuming the funding levels required to accomplish the tasks are available. The FY 2020 revision of the ATR Integrated Strategic Operations Plan dated November 11, 2029,[59] has revised the ATR core internals changeout reactor shutdown period to March 2021 through early December 2021.

A detailed cost breakdown is shown by year in Figure 4, Figure 5, and Figure 6. Fabrication, irradiation, and PIE and safety testing activities are grouped by experiment (AGR-1, AGR-2, etc.). Separate cost lines are shown for fuel performance modeling and fission product transport scopes. Additional lines are provided for the other activities described in Subsection 3.5 that cut across the program WBS elements. Costs in Figure 4, Figure 5, and Figure 6 are actual costs through FY 2015. The budget figures for FY 2018 are included, and lifecycle baseline estimates are provided for activities in FYs 2019 through 2026. In the figures, the costs are also broken down by each of the major activities in the WBS.

The AGR-5/6/7 experiment total lifecycle cost estimate is higher than the earlier AGR experiments for several reasons. First, the fuel fabrication costs are significantly greater because of the program decision made in February 2014 to fabricate new AGR-5/6/7 fuel kernels rather than use the original lot. The fuel kernels within the original lot met the fuel specifications but had fissures that were thought to fracture during the coating process and result in misshapen TRISO particle fuel. Thermal analysis of these misshapen particles demonstrated that areas of excessive stress during irradiation were likely to occur and cause the particles to fail. Fabrication of new fuel kernels for the AGR-5/6/7 experiments required the hiring of new operators and staff at BWXT-NOG with related training and qualification. The kernel fabrication equipment and processes had to be restarted, requiring maintenance and repair of equipment. This delay then caused a cascade of other delays—maintenance and repair to the coating, overcoating, and compacting equipment in order to be fully functional after an extended shutdown. Second, the AGR-5/6/7 test train design is quite different from the previous experiment designs in order to accomplish the test objectives, which has increased the costs. Also, PIE performed to date on the AGR-2 experiment identified thermocouple placement as having a possible negative effect on the TRISO fuel particle performance during irradiation. Third, the testing of moisture and air ingress on compacts during safety testing will be performed in a new furnace being developed. The development, fabrication, testing, and installation of the furnace in a suitable operating location will increase the costs associated with PIE and safety testing of the AGR-5/6/7 compacts. Fourth, the PIE and safety testing will be performed at both ORNL and INL in order to complete it in a timely manner. As a result, the costs related to performing this work at both laboratories increases. A final impact to the overall costs of the AGR-5/6/7 experiments is the additional time that is being required to complete the entire testing as a result of the extended outages planned for ATR, the reduced number of annual EFPDs available for experiment irradiation, and the related need for management and oversight of operations over the longer timeframe.

The total program cost is estimated to be ~$367M, based on completing all activities described in this technical program plan, with no constraints put on annual funding levels. If the funding levels are constrained over this period, concessions will need to be made and priorities established as to which activities will be completed and which will be deferred or cancelled. PIE safety testing and fission product transport plans are based on certain assumptions, with respect to the level of fuel performance and fission product transport model validation that the NRC will accept. If further examination and analysis are required above that which has been planned, the schedule will be extended, and costs will increase above those shown.

Fuel Development and Qualification

Activity	Start	Finish
AGR-1 - Shakedown Small Coater Fuel	**5/1/2003**	**9/15/2016**
AGR-1 Fuel Fabrication	5/1/2003	9/29/2006
AGR-1 Design and Assembly	5/1/2003	12/22/2006
AGR-1 Irradiation (Large B)	12/25/2006	11/6/2009
AGR-1 Test Train Cooldown	11/7/2009	2/28/2010
AGR-1 PIE and Safety Testing	3/1/2010	9/15/2016
AGR-2 Production Scale Coater Development	**12/2/2006**	**9/30/2020**
AGR-2 Fuel Fabrication	10/2/2006	4/9/2010
AGR-2 Design and Assembly	5/5/2008	6/11/2010
AGR-2 Irradiation (Large B)	6/23/2010	10/16/2013
AGR-2 Train Cooldown	10/17/2013	7/10/2014
AGR-2 PIE and Safety Testing	7/14/2014	9/30/2020
AGR-3/4 Fission Product Transport Data	**5/8/2006**	**3/31/2021**
AGR-3/4 Fuel Fabrication	5/3/2006	9/28/2011
AGR-3/4 Design and Assembly	3/2/2009	9/30/2011
AGR-3/4 Irradiation (NEFT)	12/14/2011	4/7/2014
AGR-3/4 Train Cooldown	4/7/2014	3/18/2015
AGR-3/4 PIE and Safety Testing	3/18/2015	3/31/2022
AGR-5/6/7 Fuel Qualification	**10/1/2008**	**6/30/2026**
AGR-5/6/7 Fuel Fabrication	10/1/2008	3/31/2017
AGR-5/6/7 Design and Assembly	5/1/2014	4/17/2017
AGR-5/6/7 Irradiation (NEFT)	2/28/2018	9/10/2020
AGR-5/6/7 Test Train Cooldown	9/11/2020	12/31/2020
AGR-5/6/7 PIE and Safety Testing	01/04/202	6/30/2026
ATR Core Internals Changeout (no irradiation)	**3/1/2021**	**12/31/2021**
Fuel Performance Code Development & Data Comparison	10/20/2017	6/30/2026
Fission Product Transport and Source Term Data Development	10/1/2003	TBD

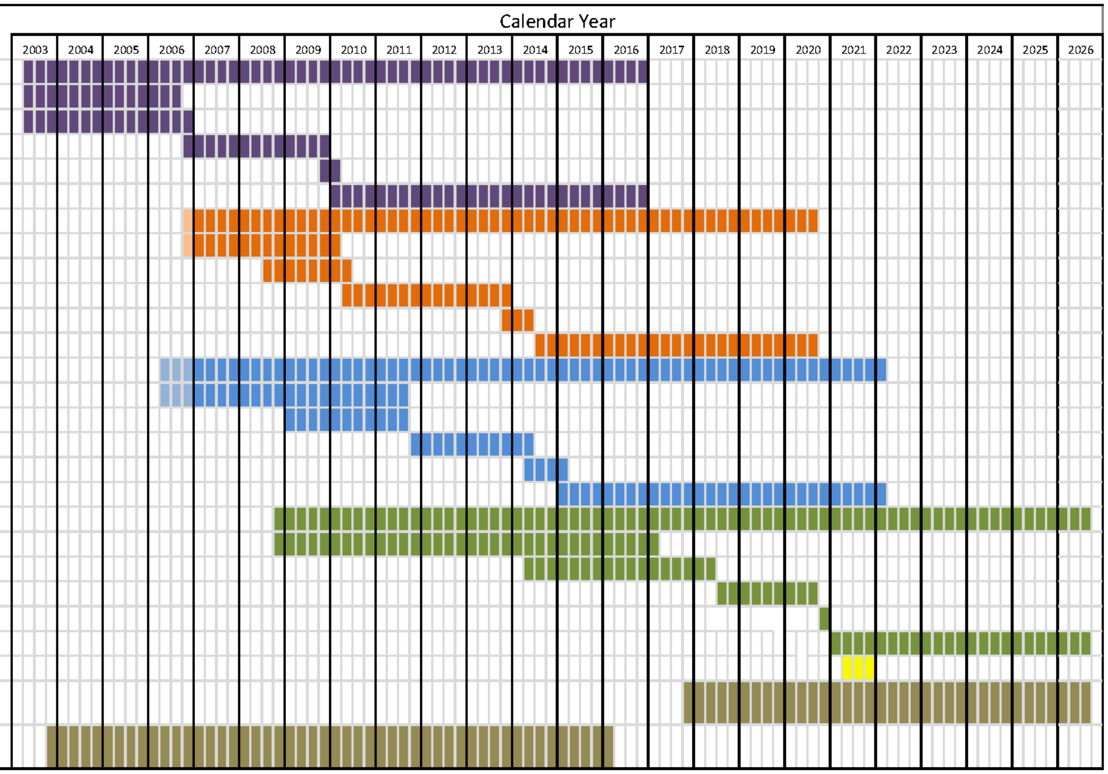

Figure 3. Fuel development and qualification higher-level summary schedule.

AGR Program	FY-03	FY-04	FY-05	FY-06	FY-07	FY-08	FY-09	FY-10
AGR-1 Shakedown Irradiation								
Fuel Fabrication	$ 450	$ 3,185	$ 2,594	$ 4,602	$ 233	$ 35	$ 67	$ 166
Design and Assembly	$ 66	$ 696	$ 62	$ 1,370	$ 108	$ (29)		
Irradiation			$ 2,252	2110	$ 1,311	$ 1,832	$ 1,745	$ 1,425
PIE				$ 215	$ 248	$ 3,101	$ 7,252	$ 9,576
Data Qualification							$ 2,256	$ 1,617
TOTAL =	**$ 516**	**$ 3,881**	**$ 4,908**	**$ 8,297**	**$ 1,900**	**$ 4,939**	**$ 11,320**	**$ 12,784**
AGR-2 Production Scale Coater								
Fuel Fabrication					$ 2,110	$ 6,660	$ 2,102	$ 767
Design and Assembly						$ 212	$ 1,231	$ 2,018
Irradiation								$ 368
PIE								
Data Qualification								$ 337
TOTAL =	**$ -**	**$ -**	**$ -**	**$ -**	**$ 2,110**	**$ 6,872**	**$ 3,333**	**$ 3,491**
AGR-3/4 Fission Product Trans								
Fuel Fabrication					$ 350	$ 206	$ 187	$ 1,095
Design and Assembly			$ 685	120	$ 4	$ 5	$ 67	$ 617
Irradiation							$ 118	$ 534
PIE								
Data Qualification								
TOTAL =	**$ -**	**$ -**	**$ 685**	**$ 120**	**$ 354**	**$ 212**	**$ 372**	**$ 2,246**
AGR-5/6/7 Fuel Qualification								
Fuel Fabrication							$ 6,608	$ 6,333
Design and Assembly								
Irradiation								
PIE								
Data Qualification								
TOTAL =	**$ -**	**$ -**	**$ -**	**$ -**	**$ -**	**$ -**	**$ 6,608**	**$ 6,333**
Fuel Performance Modeling	$ 148	$ 371	$ 710	$ 620	$ 178	$ 661	$ 1,192	$ 1,256
Fission Product Transport		$ 82	$ 46	$ 71	$ 53	$ 396	$ 714	$ 736
NRC Reports								$ -
Fuel Fab Commercialization								$ -
Facility Upgrades						$ 2,309	$ 3,811	$ 1,527
Licensing Support								
NDMAS Upgrades								$ 1,545
PM Oversight	$ 592	$ 937	$ 1,077	$ 1,433	$ 645	$ 1,648	$ 1,331	$ 1,557
SUBTOTAL =	**$ 740**	**$ 1,389**	**$ 1,833**	**$ 2,124**	**$ 876**	**$ 5,014**	**$ 7,048**	**$ 6,621**
GRAND TOTAL =	**$ 1,256**	**$ 5,270**	**$ 7,426**	**$ 10,541**	**$ 5,240**	**$ 17,037**	**$ 28,682**	**$ 31,475**
Cumulative actual total	$ 1,256	$ 6,526	$ 13,952	$ 24,493	$ 29,733	$ 46,770	$ 75,451	$ 106,926

FY03-FY18 Total Actuals

FY19 Estimated Costs

FY20-FY26 Projected Costs based on scheduled activities (Includes PM Oversight and Technical Integration)

Figure 4. Fuel development and qualification annual costs for FYs 2003 through 2010.

AGR Program	FY-11	FY-12	FY-13	FY-14	FY-15	FY-16	FY-17	FY-18	FY-19
AGR-1 Shakedown Irradiation									
Fuel Fabrication	$ 102	$ 23							
Design and Assembly									
Irradiation	$ 2	$ 248	61						
PIE	$ 6,549	$ 5,165	$ 5,901	$ 4,936	$ 1,809	$ 1,636	$ 203		
Data Qualification	$ 254	$ 175	43	$ 80	$ 215	$ 144	$ 182		
TOTAL =	**$ 6,907**	**$ 5,611**	**$ 6,005**	**$ 5,016**	**$ 2,024**	**$ 1,780**	**$ 385**		
AGR-2 Production Scale Coater									
Fuel Fabrication									
Design and Assembly	$ 3								
Irradiation	$ 2,624	$ 1,263	$ 1,106	$ 743	$ 2				
PIE		$ 41	$ 305	$ 870	$ 2,832	$ 5,110	$ 4,268	$ 5,075	$ 4,119
Data Qualification	$ 1,053	$ 1,081	$ 212	$ 123	$ 279	$ 131	$ 91	$ 15	$ 34
TOTAL =	**$ 3,680**	**$ 2,385**	**$ 1,623**	**$ 1,736**	**$ 3,113**	**$ 5,241**	**$ 4,359**	**$ 5,090**	**$ 4,153**
AGR-3/4 Fission Product Trans									
Fuel Fabrication	$ 1,948	$ 246							
Design and Assembly	$ 3,499	$ 37							
Irradiation	$ 1,792	$ 2,757	$ 3,003	$ 2,468	$ 824				
PIE		$ 544	$ 583	$ 549	$ 1,487	$ 1,852	$ 4,143	$ 3,761	$ 2,704
Data Qualification	$ 91	$ 73	$ 607	$ 326	$ 450	$ 392	$ 355	$ 78	$ -
TOTAL =	**$ 7,330**	**$ 3,657**	**$ 4,193**	**$ 3,343**	**$ 2,761**	**$ 2,244**	**$ 4,498**	**$ 3,839**	**$ 2,704**
AGR-5/6/7 Fuel Qualification									
Fuel Fabrication	$ 6,881	$ 4,558	$ 3,323	$ 2,910	$ 4,197	$ 3,833	$ 3,585	$ 908	$ 1,032
Design and Assembly			$ 466	$ 860	$ 2,280	$ 2,430	$ 1,979		
Irradiation					$ -	$ -	$ -	$ 1,687	$ 1,481
PIE									$ 190
Data Qualification					$ 6	$ 4	$ 46	$ 332	$ 391
TOTAL =	**$ 6,881**	**$ 4,558**	**$ 3,789**	**$ 3,770**	**$ 6,483**	**$ 6,267**	**$ 5,610**	**$ 2,927**	**$ 3,094**
AMIX Oxidation Testing									
Design and Assembly					$ 31	$ 452	$ 609	$ 3,135	$ 1,693
Oxidation Testing									
TOTAL =	**$ -**	**$ -**	**$ -**	**$ -**	**$ 31**	**$ 452**	**$ 609**	**$ 3,135**	**$ 1,693**
Fuel Performance Modeling	$ 611	$ 758	$ 610	$ 455	$ 530	$ 1,029	$ 651	$ 546	$ 520
Fission Product Transport	$ 641	$ 174							
NRC Reports									
Fuel Fab Commercialization									
Facility Upgrades	$ 1,568	$ 836			$ 435	$ -	$ 488		
Licensing Support							$ 2	$ 152	$ 162
NDMAS Upgrades	$ 1,353	$ 597	$ 262	$ 642	$ 659	$ 651	$ 744	$ 706	$ 550
PM Oversight	$ 1,639	$ 1,289	$ 736	$ 816	$ 1,567	$ 1,933	$ 1,650	$ 1,416	$ 1,367
SUBTOTAL =	**$ 5,812**	**$ 3,654**	**$ 1,608**	**$ 1,913**	**$ 3,191**	**$ 3,613**	**$ 3,535**	**$ 2,820**	**$ 2,599**
GRAND TOTAL =	**$ 30,610**	**$ 19,865**	**$ 17,218**	**$ 15,778**	**$ 17,603**	**$ 19,597**	**$ 18,996**	**$ 17,811**	**$ 14,243**
Cumulative actual total	$ 137,537	$ 157,401	$ 174,619	$ 190,397	$ 208,000	$ 227,597	$ 246,593	$ 264,404	$ 278,647
FY03-FY19 Total Actuals									
FY20 Estimated Costs									
FY21-FY26 Projected Costs based on scheduled activities (Includes PM Oversight and Technical Integration)									

Figure 5. Fuel development and qualification annual costs for FYs 2011 through 2019.

AGR Program	FY-20	FY-21	FY-22	FY-23	FY-24	FY-25	FY-26	FY-27	Total
AGR-1 Shakedown Irradiation									
Fuel Fabrication									$ 11,457
Design and Assembly									$ 2,273
Irradiation									$ 10,987
PIE									$ 46,591
Data Qualification									$ 4,966
TOTAL =									**$ 76,274**
AGR-2 Production Scale Coater									
Fuel Fabrication									$ 11,640
Design and Assembly									$ 3,464
Irradiation									$ 6,106
PIE	$ 2,826	$ 1,050							$ 26,496
Data Qualification	$ 75	$ 75							$ 3,506
TOTAL =	**$ 2,901**	**$ 1,125**							$ 51,212
AGR-3/4 Fission Product Trans									
Fuel Fabrication									$ 4,033
Design and Assembly									$ 5,034
Irradiation									$ 11,496
PIE	$ 4,250	$ 4,000	$ 1,250						$ 25,123
Data Qualification	$ 75	$ 100	$ 125						$ 2,672
TOTAL =	**$ 4,325**	**$ 4,100**	**$ 1,375**						**$ 48,358**
AGR-5/6/7 Fuel Qualification									
Fuel Fabrication	$ 1,130	$ 250							$ 45,548
Design and Assembly									$ 8,015
Irradiation	$ 1,850	$ 1,200							$ 6,218
PIE	$ 975	$ 3,250	$ 6,450	$ 7,400	$ 7,400	$ 7,400	$ 3,300		$ 36,365
Data Qualification	$ 385	$ 300	$ 400	$ 525	$ 525	$ 525	$ 400		$ 3,839
TOTAL =	**$ 4,340**	**$ 5,000**	**$ 6,850**	**$ 7,925**	**$ 7,925**	**$ 7,925**	**$ 3,700**		**$ 99,985**
AMIX Oxidation Testing									
Design and Assembly	$ 1,900	$ 2,050							$ 9,870
Oxidation Testing		$ -	$ 3,200	$ 3,400	$ 3,400	$ 3,400	$ 1,500		$ 14,900
TOTAL =	**$ 1,900**	**$ 2,050**	**$ 3,200**	**$ 3,400**	**$ 3,400**	**$ 3,400**	**$ 1,500**		**$ 24,770**
Fuel Performance Modeling	$ 320	$ 750	$ 400	$ 400	$ 400	$ 400	$ -		$ 13,516
Fission Product Transport									$ 2,913
NRC Reports									$ -
Fuel Fab Commercialization									$ -
Facility Upgrades									$ 10,974
Licensing Support	$ 233	$ 100	$ 100	$ 200	$ 200	$ 200	$ 200		$ 1,549
NDMAS Upgrades	$ 560	$ 575	$ 575	$ 575	$ 575	$ 575	$ 400		$ 11,544
PM Oversight	$ 1,206	$ 1,500	$ 1,500	$ 1,500	$ 1,500	$ 1,500	$ 1,200		$ 31,539
SUBTOTAL =	**$ 2,319**	**$ 2,925**	**$ 2,575**	**$ 2,675**	**$ 2,675**	**$ 2,675**	**$ 1,800**		**$ 72,034**
GRAND TOTAL =	**$ 15,785**	**$ 15,200**	**$ 14,000**	**$ 14,000**	**$ 14,000**	**$ 14,000**	**$ 7,000**	**$ -**	**$ 372,632**
Cumulative actual total	$ 294,432	$ 309,632	$ 323,632	$ 337,632	$ 351,632	$ 365,632	$ 372,632		
FY03-FY19 Total Actuals									
FY20 Estimated Costs									
FY21-FY26 Projected Costs based on scheduled activities (Includes PM Oversight and Technical Integration)									

Figure 6. Fuel development and qualification annual costs for FYs 2020 through 2026.

5. REFERENCES

1. S. Muñoz, "Fuel Product Specification," DOE-HTGR-100209, May 1994.
2. ORNL, *Technical Program Plan for the Advanced Gas Reactor Fuel Development and Qualification Program*, ORNL/TM-2002/262, April 2003.
3. Idaho National Laboratory, *NGNP Fuel Qualification White Paper*, INL/EXT-10-17686, Rev. 0, May 22, 2010.
4. Wayne Moe, *Mechanistic Source Terms White Paper*, INL/EXT-10-17997, Rev. 0, Idaho National Laboratory, July 21, 2010.
5. Glenn M. Tracy, NRC, Office of New Reactors, to Dr. John E. Kelly, DOE-NE, "Next Generation Nuclear Plant—Assessment of Key Licensing Issues," July 17, 2014.
6. William F. Martin, Chairman, and John Ahearne, Vice Chairman, Nuclear Energy Advisory Committee, letter to Dr. Steven Chu, Secretary of Energy, June 30, 2011.
7. G. Bourchardt, B. Hürttlen, and G. Pott, *Experiment FRJ2-P24 Bestrahlungsbericht, Kernsforschugsanlage Jülich GMBH*, Report No. ZBB/IB/19/82.
8. W. Schenk, R. Gontard, and H. Nabielek, P*erformance of HTR Fuel Samples under High-Irradiation and Accident Simulation Conditions, with Emphasis on Test Capsules HFR-P4 and SL-P1*, Research Center Jülich, October 1994.
9. R. R. Hobbins and R. K. McCardell, *Summary of NP-MHTGR Fuel Failure Evaluation*, EGG-NPR-10967, September 1993.
10. General Atomics, *MHTGR TRISO-P Fuel Failure Evaluation Report*, DOE-HTGR-90390, Rev. 0, October 1993.
11. EDF-4198, "Preliminary AGR Fuel Specification," Rev. 1, Idaho National Laboratory, April 2004.
12. B. P. Collin, *AGR-1 Irradiation Test Final As-Run Report*, INL/EXT-10-18097, Rev. 3, January 2015.
13. B. P. Collin, *AGR-2 Irradiation Test Final As-Run Report*, INL/EXT-14-32277, Rev. 3, February 2018.
14. B. P. Collin, *AGR-3/4 Irradiation Test Final As-Run Report*, INL/EXT-15-35550, Rev. 1, May 2016.
15. B. Collin, *AGR-1 Safety Test Predictions Using the PARFUME Code*, INL/EXT-12-26014, May 2012.
16. B. Collin, *AGR-2 Safety Test Predictions Using the PARFUME Code*, INL/EXT-14-33082, September 2014.
17. B. T. Pham, J. J. Einerson, and G. L. Hawkes, *Uncertainty Quantification of Calculated Temperatures for AGR-3/4 Experiment*, INL/EXT-15-36431, Rev. 1, April 2016.

18. I. J. van Rooyen, T. M. Lillo, H. Wen, K. E. Wright, J. Madden, and J. A. Aguiar, *Advanced Electron Microscopy and Micro analytical Technique Development and Application for Irradiated TRISO Coated Particles from the AGR-1 Experiment*, INL/EXT-15-36281, Rev 0, January 2017.

19. P. A. Demkowicz, *AGR-1 Post Irradiation Examination Final Report*, INL/EXT-15-36407, August 2015.

20. J.D. Stempien, M. Plummer, J. Schulthess, *Measurement of Kernel Swelling and Buffer Densification in Irradiated UCO and UO_2 TRISO Fuel Particles from AGR-2*, INL/EXT-19-54502, June 2019.

21. J.D. Hunn, F.C. Montgomery, T.J. Keever, B.D. Roach, R.H. Ilgner, *Determination of Burnup in AGR-2 Compacts 3-3-1, 3-3-2, 2-2-2, and 6-4-2*, ORNL/TM-2019/1315, September 2019.

22. S. Meher, C. Jiang, I.J. van Rooyen, "Understanding of Fission Products Transport in SiC Layer of TRISO Fuels by Nanoscale Characterization and Modeling," *Journal of Nuclear Materials* 527 (2019), 151793.

23. C. Jiang, I.J. van Rooyen, S. Meher, "*Ab Initio* Study and Thermodynamic Modeling of the Pd-Si-C System," *Computational Materials Science* 171 (2020) 109238,

24. T.J. Gerczak, C. Contescu, J.J. Lee, R. Mee, A. Schumacher, J. Stempien, M. Trammell, *Oxidation of Matrix Material in Helium with Varied Moisture Content*, ORNL/TM-2019/1341, November 2019.

25. J. D. Stempien, F. J. Rice, P. L. Winston, and J. Harp, *AGR-3/4 Irradiation Test Train Disassembly and Component Metrology First Look Report*, INL/EXT-16-38005, Rev 1, September 2016.

26. G. L. Hawkes, "AGR-3/4 Daily As-Run Thermal Analyses," ECAR-2807, Rev. 1, May 2016.

27. J. D. Stempien and J. Schulthess, *AGR-3/4 TRISO Fuel Compact Ceramography*, INL/EXT-20-57610, March 2020.

28. B. P. Collin, D. A. Petti, P. A. Demkowicz, J. T. Maki, "Comparison of silver, cesium, and strontium release predictions using PARFUME with results from the AGR-1 irradiation experiment," *Journal of Nuclear Materials* 466 (2015) 426-442 (INL/JOU-15-35343).

29. B. P. Collin, D. A. Petti, P. A. Demkowicz, J. T. Maki, "Comparison of fission product release predictions using PARFUME with results from the AGR-1 safety tests," *Nuclear Engineering and Design* 301 (2016) 378-390 (INL/JOU-15-34147).

30. W.F. Skerjanc, J.T. Maki, B.P. Collin, D.A. Petti, "Evaluation of design parameters for TRISO-coated fuel particles to establish manufacturing critical limits using PARFUME," *Journal of Nuclear Materials* Vol. 469, pp. 99–105, 2016.

31. G.K. Miller, D.A. Petti, J.T. Maki, D.L. Knudson, W.F. Skerjanc, *PARFUME Theory and Model Basis Report*, INL/EXT-08-14497, Rev. 1, September 2018.

32. W.F. Skerjanc, *PARFUME User's Guide*, INL/EXT-10-18317, Rev. 1, September 2019.

33. W.F. Skerjanc and B.P. Collin, *Assessment of Material Properties for TRISO Fuel Particles used in PARFUME*, INL/EXT-18-44631, Rev. 0, August 2018.

34. Idaho National Laboratory, *NGNP Fuel Qualification White Paper*, INL/EXT-10-18610, Rev. 0, July 2010.
35. R. R. Hobbins, "Research Plan for Moisture and Air Ingress Experiments," PLN-4086, Rev. 1, April 2012.
36. NUREG-1338, "Draft Preapplication Safety Evaluation Report for the Modular High-Temperature Gas-Cooled Reactor," March 1989.
37. NUREG-1338, "Preapplication Safety Evaluation Report for the Modular High-Temperature Gas-Cooled Reactor (MHTGR),"*)*", Draft Copy of the Final Report, December 1995.
38. EPRI, "Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO) Coated Particle Fuel Performance," Topical Report EPRI-AR-1 (NP), 2019.
39. 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," Appendix B – Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants, Code of Federal Regulations, January 1, 2008.
40. ASME NQA-1-2008/1a-2009, "Quality Assurance Requirements for Nuclear Facility Applications.
41. R. Gontard and H. Nabielek, *Performance Evaluation of Modern HTR TRISO Fuels*, HTA-IB-05/90, July 1990.
42. D. A. Petti, J.T. Maki, J. Buongiorno, R.R. Hobbins, G.K. Miller, *Key Differences in the Fabrication, Irradiation, and Safety Testing of U.S. and German TRISO particle fuel and Their Implications on Fuel Performance*, INEEL/EXT-02-00300, June 2002.
43. J. Ketterer, *Capsule HRB-16 Pre-irradiation Report*, General Atomics 906200, Issue 1, September 1981.
44. General Atomics, *Capsule HRB-21 Pre-irradiation Report*, DOE-HTGR-88357, Rev. C, April 1991.
45. Babcock & Wilcox, *NPR-MHTGR Performance Test Fuel, Uranium Oxycarbide Fuel Kernels Composite, Lot Number: B10-K-91381*, April 26, 1991.
46. EDF-4380, "AGR-1 Fuel Product Specification and Characterization Guidance," Rev. 8, Idaho National Laboratory, January 2011.
47. R. Acharya, C. A. Baldwin, G.L. Bell, L.C. Emerson, M.L. Grossbeck, F.C. Montgomery, R.N. Morris, B.F. Myers, N.H. Packan, J.T. Parks, O.M. Stansfield, *Fuel Capsule HRB-21 Post-irradiation Examination Data Report*, DOE-HTGR-100229, April 1995.
48. P. A. Demkowicz, L. Cole, S. Ploger, and P. Winston, 2011, *AGR 1 Irradiated Test Train Preliminary Inspection and Disassembly First Look*, INL/EXT-10-20722, Rev. 0, Idaho National Laboratory, January 2010.
49. S. Ploger, P. Demkowicz, J. Harp, *AGR-2 Irradiated Test Train Preliminary Inspection and Disassembly First Look*, INL/EXT-15-34997, May 2015.
50. J. D. Stempien, "Furnace for Testing TRISO Fuels Under Air and Moisture-Ingress Accident Conditions," FOR-284, Idaho National Laboratory, August 2016.

51. D. G. Martin, “Physical and Mechanical Properties of the Constituents of Coated Particles and the Effect of Irradiation,” *HTR-F WP3 Meeting, Lyon, France*, October 18, 2001.

52. International Atomic Energy Agency, *Advances in High Temperature Gas Cooled Reactor Fuel Technology*, IAEA-TECDOC-1674, 2012.

53. S. Knol, S. de Groot, P. R. Hania, M. H.C. Hannink, “PYCASSO: Irradiation of HTR coated particles at high temperatures,” Nuclear Engineering and Design 251 (2012) 150-156.

54. C. Folsom, C. Xing, C. Jensen, H. Ban and D. Marshall, “Experimental Measurement and Numerical Modeling of the Effective Thermal Conductivity of TRISO Fuel Compacts,” *Journal of Nuclear Materials* 458 (2015), pp., 198-205.

55. K. D. Hamman, “AGR -2 Pre-Test Prediction Analyses using the PARFUME Code for the U.S. Fuel Particles,” ECAR-1020, Rev. 2, May 2012.

56. W. F. Skerjanc and B. Collin, *AGR-3/4 Irradiation Test Predictions using PARFUME*, INL/EXT-16-38280, March 2016.

57. P. Calderoni and M. A. Ebner, *Hydrogen Permeability of Incoloy 800H, Inconel 617, and Haynes 230 Alloys*, INL/EXT-10-19387, July 2010.

58. P. Humrickhouse, R. Pawelko, M. Shimada, and P. Winston, *Tritium Permeability of Incoloy 800H and Inconel 617*, INL/EXT-11-23265, Rev. 1, July 2012.

59. ATR-TPS-11-11-19, “FY-20 Integrated Strategic Operational Plan (ISOP),” Revision 0a, November 2019.