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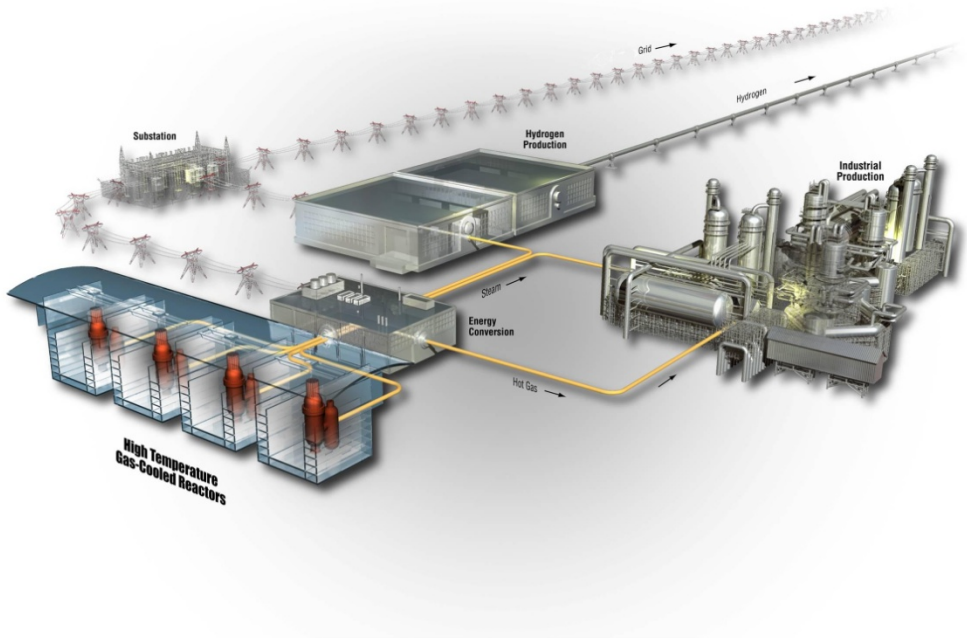
Plan

Technical Program Plan for the Next Generation Nuclear Plant/Advanced Gas Reactor Fuel Development and Qualification Program



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TECHNICAL PROGRAM PLAN FOR THE NEXT GENERATION NUCLEAR PLANT/ADVANCED GAS REACTOR FUEL DEVELOPMENT AND QUALIFICATION PROGRAM	Identifier: PLN-3636 Revision: 0 Effective Date: 09/30/10	Page: i of xii

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Prepared by:

Jack Simonds — 9/27/2010

J. Simonds
Project Manager

Date

Approved by:

Dave Petti — 9/27/2010

D. A. Petti
VHTR TDO Technical Director

Date

Gary D Roberts — 9/27/2010

G. D. Roberts
Quality Assurance

Date

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SUMMARY

High temperature gas-cooled reactors (HTGRs) are graphite-moderated nuclear reactors cooled by helium. The high outlet temperatures and high thermal-energy conversion efficiency of HTGRs enable an efficient and cost-effective integration with non-electricity generation applications, such as process heat and hydrogen production, for the many petrochemical and other industrial processes that require temperatures between 300 and 900°C. Using HTGRs in this way would 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 CO_2 emissions. Key characteristics of this reactor design are the use of helium as a coolant, graphite as the moderator of neutrons, and ceramic-coated particles as fuel. Helium is chemically inert and neutronically transparent. The graphite core slows down the neutrons and provides high-temperature strength and structural stability for the core and a substantial heat sink during transient conditions. The ceramic-coated particle fuel is extremely robust and retains the radioactive byproducts of the fission reaction under both normal and off-normal conditions.

The U.S. Department of Energy (DOE) has selected the HTGR for the Next Generation Nuclear Plant (NGNP) Project as a transformative application of nuclear energy that will demonstrate emissions-free nuclear-assisted electricity, process heat, and hydrogen production, thereby reducing greenhouse-gas emissions and enhancing energy security. The first of a kind HTGR envisioned for the NGNP is an extension of past applications of gas-cooled reactor technologies and will be driven by near-term commercial industry needs and current technology availability. The NGNP reference concept will be an HTGR with a design-goal outlet gas temperature of 750 to 800°C. The reactor core could be either a prismatic graphite-block core or a pebble bed core. Deployment options supported by the NGNP will be able to produce electricity, process heat, and hydrogen. The fuel cycle will be a once-through, high-burnup, low-enriched uranium fuel cycle.

Two HTGR power reactors (115-MW(t) Peach Bottom Unit 1 and 842-MW(t) Fort St. Vrain) were built and operated in the U.S. in the 1960s through the 1980s. In the 1980s, increasing recognition of the organizational and operational complexities arising from reliance on powered, active safety systems fostered the development of modular HTGR-based concepts, which emphasized inherent passive features to meet public safety requirements. Two basic design options currently exist for HTGRs: the pebble bed reactor (PBR) and the prismatic modular reactor (PMR). The PBR uses fuel particles formed into tennis ball-size graphite spheres that slowly move through the reactor core and are recycled or replaced in a continuous refueling process. The PMR uses fuel particles formed into cylindrical rods that are loaded into hexagonal-shaped graphite blocks. These blocks make up the reactor core, which is periodically refueled in a batch process. Modern conceptual designs of both concepts place greater reliance on retention of fission products in coated-particle fuel, resulting in higher standards of quality and performance than were required of fuel for earlier HTGR plants.

The objective of the DOE Advanced Gas Reactor (AGR) Fuel Development and Qualification program is to qualify tristructural-isotropic (TRISO)-coated particle fuel for use in HTGRs. The NGNP/AGR Fuel Development and Qualification Program was established to achieve the following overall goals:

- Provide a fuel qualification data set in support of the licensing and operation of the NGNP. Gas-cooled reactor fuel performance demonstration and qualification comprise the longest duration research and development task required for NGNP design and licensing. The fuel form is to be demonstrated and qualified for service conditions enveloping normal operation and accidents.

- Support deployment of the NGNP for hydrogen 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 resources by using international collaboration mechanisms.

These goals are being addressed by conduct of coated-particle fuel development and qualification activities that support both the PMR and PBR fuel designs at different levels of effort. A broad body of international data is applicable to the PBR concepts; additional development has been under way in China, South Africa, Japan, and the European Union over the last decade. Because these results are available to the U.S. via international collaboration, less focus has been on pebble bed fuel. Furthermore, while these results are relevant for PMR designs, these data are not sufficient to address specific prismatic-core manufacturing parameters and the typically higher PMR fuel service condition requirements (higher burnup, operating temperature and temperature gradients, and fast neutron exposure). Thus, early effort in the AGR program has been focused on manufacturing and testing the fuel design used in the PMR concepts, using the most recent gas turbine modular helium reactor fuel product specification[a] as a starting point. Irradiation, safety testing, and post-irradiation examination (PIE) plans are based on supporting fuel development and qualification in an integrated manner. If a PBR design should be selected for the NGNP, the irradiation and PIE activities would be expanded or redirected, as necessary, starting with a preliminary plan developed for PBR fuel qualification.[b] However, because international pebble bed TRISO fuel programs have been recently curtailed, more work would need to be done under the AGR program to augment the existing German fuel performance database for a pebble bed NGNP.

Preliminary operating conditions and performance requirements for the fuel and preliminary fuel product specifications to guide the program's fuel process development activities have been generated based on previously completed HTGR design and technology development activities, operating conditions and performance requirements.

At the onset of the program in 2002, facilities and personnel experienced in the activities necessary to address the program goals existed in the U.S., primarily at Idaho National Laboratory (INL) and Oak Ridge National Laboratory (ORNL). INL and ORNL personnel with the relevant experience and knowledge of TRISO fuel, facility status and capabilities were directly involved in developing the initial Technical Program Plan. In addition, General Atomics provided input regarding PMR fuel performance requirements and perspectives from its prior experience in fuel development, fuel fabrication, and fuel-related analytical capabilities needed to support licensing interactions. Babcock and Wilcox also provided input based on its experience and capabilities for fuel kernel production and particle coating. Many of the individuals who developed this plan were directly involved in the production and testing of previous U.S. fuel for the Modular High Temperature Gas Reactor and the New Production Reactor, as well as the extensive investigations and reviews conducted in the early 1990s following the unexpectedly high failure levels observed in those tests. Thus, the plan builds directly on the large body of coated-particle fuel experience and is generally consistent with the recommendations arising from that experience.

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a. DOE-GT-MHR-100209, Fuel Product Specification, May 1994.
b. PBMR Fuel Qualification Test Program, March 2002, NRC Adams Accession No. ML020800192.NGNP Fuel Qualification White Paper, INL/EXT-10-18610, Revision 0, July 2010.

The NGNP/AGR Fuel Development and Qualification Program involves five major program elements:

1. *Fuel fabrication.* The path to successful coated-particle fuel fabrication (manufacturing fuel that meets the fuel quality and performance requirements for the NGNP) is to develop a coating process that replicates in the NGNP particle design, to the greatest extent possible, the properties of the coatings on German fuel particles that have exhibited superior irradiation and accident performance. Coating process development is accomplished in two phases: the first being conducted in a 2-inch diameter laboratory coater and the second scaling up to a 6-inch prototypic production-size coater. The fuel fabrication program element also includes establishing the fuel fabrication infrastructure, process development for the kernels and compacts, coating process model development, quality control (QC) methods development, fuel process scale-up analyses, and development of the process documentation needed for technology transfer to industry. The fuel fabrication effort will produce coated-particle fuel that meets fuel product and provides fuel and material samples for characterization, irradiation, and safety testing as necessary to meet the overall goals. The plan also identifies work to develop automated fuel fabrication technology suitable for mass production of coated-particle fuel at an acceptable cost.

2. *Fuel and material irradiation.* Fuel and material irradiation activities will provide data on fuel performance under 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. It will also provide irradiated fuel and materials as necessary for PIE and safety testing. A total of eight irradiation tests, designated as AGR-1 through AGR-8, have been defined to provide the necessary data and sample materials.

3. *Fuel PIE and safety testing.* This program element will provide the facilities and processes to measure the performance of AGR fuel systems under normal operating conditions and accident conditions. This work will support the fuel manufacturing effort by providing feedback on the performance of kernels, coatings, and compacts. PIE and safety testing will provide a broad range of data on fuel performance and fission product transport within 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 ratio) are necessary to demonstrate compliance with fuel performance requirements and support the development and validation of computer codes.

4. *Fuel performance modeling.* Fuel performance modeling, as defined in the context of this plan, addresses the structural, thermal, and chemical processes that can lead to coated-particle failures. It considers the effect 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 is also modeled. Computer codes and models will be further developed and validated as necessary to support fuel fabrication process development and plant design and licensing.

5. *Fission product transport and source term.* This element will address the transport to the exclusion area boundary of fission products produced in the coated particles to provide a technical basis for source terms for AGRs under normal and accident conditions. The technical basis will be codified in design methods (computer models) validated by experimental data, as needed, to support plant design and licensing.

An underlying theme for the work scope presented in this plan is the need to develop an understanding of the relationship between the fuel fabrication process, fuel product properties, and irradiation and safety testing performance. Precision process control, along with advanced characterization and data acquisition methods, conducted within a structured quality assurance framework are important elements of achieving this objective. Production of qualified data on 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 also important elements. Fuel performance modeling and analysis of fission product behavior in the primary circuit and in the reactor building are also included in this program plan. The performance models are considered essential for several reasons, including (a) guidance for the plant designer in establishing the core design and operating limits, and (b) demonstration to the licensing authority that the applicant has a thorough understanding of the in-service behavior of the fuel system. The fission product behavior task will provide primary source term data needed for licensing.

The five program elements and the activities associated with each are discussed in Section 3 of this plan. Current status and key accomplishments to date are summarized in parallel subsections in Section 5. Early program activities were centered on the fuel manufacture element, because the production of fuel and materials for irradiation, safety testing, and PIE is the early critical path activity. Other important early activities have been the design of the irradiation test train for the first irradiation and the establishment of fuel performance analysis capability to provide pretest predictions. Lower levels of activity in the other elements are associated with defining the required test articles and irradiation conditions, establishing specific plans for providing the necessary capabilities, and addressing long lead activities. In the development of the original version of the technical program plan, priority was given to the early activities in support of near-term execution of the plan. Issues associated with longer-term activities are being addressed in more detail as they arise during execution of the plan, and their impact is being factored into the overall planning. This additional detail has not affected the basic logic of the plan, but does affect the details of its execution. This plan will continue to be periodically updated to reflect additional knowledge and the results of ongoing and completed work. This revision is the third of such updates. Plan execution will be adjusted according to funding changes and limitations, in terms of milestones, completion dates, and scope. Annual implementation plans will be issued based on the actual funding received and changes in technical directions as they evolve.

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ACRONYMS

ADUN	acid-deficient uranyl nitrate
AGR	advanced gas reactor
ATR	Advanced Test Reactor (INL)
AVR	Arbeitsgemeinschaft Versuchsreaktor (Jülich, Germany)
B&W	Babcock and Wilcox
BISO	bistructural isotropic (fuel particles)
BWXT	BWX Technologies
CCCD	core conduction cooldown
DDN	design data need
DOE	Department of Energy
DTF	designed-to-fail
FIMA	fissions per initial metal atom
FSV	Fort St. Vrain
FY	fiscal year
GA	General Atomics
GIF	Generation IV International Forum
GT-MHR	gas turbine modular helium reactor
HFEF	Hot Fuel Examination Facility (INL)
HFIR	High Flux Isotope Reactor (ORNL)
HFR	High Flux Reactor (Petten, Netherlands)
HMTA	hexamethylene tetramine
HRB	high flux isotope reactor removable beryllium
HTGR	high temperature gas-cooled reactor
HTR	High Temperature Reactor (China) or High-Temperature Reactor (Germany)
HTTR	High Temperature Test Reactor (Japan)
IAEA	International Atomic Energy Agency
INL	Idaho National Laboratory
IPyC	inner pyrolytic carbon
JAEA	Japan Atomic Energy Agency (formerly JAERI)
LBL	leach-burn-leach

LEU	low enriched uranium
LEUCO	low enriched uranium oxycarbide
LOMI	low oxidation-state metal ion
LWR	light water reactor
MGEM	Modulator Generalized Ellipsometry Microscope
MHTGR	Modular High Temperature Gas Reactor
MTS	methyltrichlorosilane
NERI	Nuclear Energy Research Initiative
NGNP	Next Generation Nuclear Plant
NPR	New Production Reactor
NQA	Nuclear Quality Assurance
NRC	Nuclear Regulatory Commission
NUCO	natural uranium oxycarbide
OPyC	outer pyrocarbon
ORNL	Oak Ridge National Laboratory
PARFUME	Particle Fuel Model
PBMR	Pebble Bed Modular Reactor (Pty) Limited
PBR	pebble bed reactor
PIE	post-irradiation examination
PMR	prismatic modular reactor
PSID	Preliminary Safety Information Document
QA	quality assurance
QC	quality control
R&D	research and development
R/B	release rate-to-birth rate
RCRA	Resource Conservation and Recovery Act
SEM	scanning electron microscope
SiC	silicon carbide
SR	shear ratio
TCE	trichloroethylene
TRIGA	training, research, and isotope reactor (built by GA)
TRISO	tristructural-isotropic (fuel particles)

UCO	uranium oxycarbide (fuel)
V&V	verification and validation
VHTR	very high temperature reactor
VLPC	vented low-pressure containment
WBS	work breakdown structure

1. INTRODUCTION

High Temperature Gas-cooled Reactors (HTGRs) are graphite-moderated nuclear reactors cooled by helium. The high outlet temperatures and high thermal-energy conversion efficiency of HTGRs enable an efficient and cost-effective integration with non-electricity generation applications, such as process heat and/or hydrogen production, for the many petrochemical and other industrial processes that require temperatures between 300 and 900°C. Using HTGRs in this way would 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 CO_2 emissions. Key characteristics of this reactor design are the use of helium as a coolant, graphite as the moderator of neutrons, and ceramic-coated particles as fuel. Helium is chemically inert and neutronically transparent. The graphite core slows down the neutrons and provides high-temperature strength and structural stability for the core and a substantial heat sink during transient conditions. The ceramic-coated particle fuel is extremely robust and retains the radioactive byproducts of the fission reaction under both normal and off-normal conditions.

The U.S. Department of Energy (DOE) has selected the HTGR for the Next Generation Nuclear Plant (NGNP) Project as a transformative application of nuclear energy that will demonstrate emissions-free nuclear-derived electricity, process heat, and hydrogen production, thereby reducing greenhouse-gas emissions and enhancing energy security. The first of a kind HTGR envisioned for the NGNP is an extension of past applications of gas-cooled reactor technologies and will be driven by near-term commercial industry needs and current technology availability. The NGNP reference concept will be an HTGR with a design-goal outlet gas temperature of 750 to 800°C. The reactor core could be either a prismatic graphite-block core or a pebble bed core. Deployment options supported by the NGNP will be able to produce electricity, process heat, and hydrogen. The fuel cycle will be a once-through, high-burnup, low-enriched uranium fuel cycle.

In fiscal year (FY) 2002, the Office of Nuclear Energy, Science, and Technology initiated the development of a plan for an advanced gas reactor (AGR) fuel development and qualification program for coated-particle fuel. The resulting *Technical Program Plan for an Advanced Gas Reactor Fuel Development and Qualification Program*[1] defines fuel development activities to support the licensing and operation of a high temperature gas-cooled reactor system such as the NGNP in the U.S.

This revision describes the updated plan for the development and qualification of fuel for the NGNP, which incorporates the experience and knowledge gained from ongoing and completed work. This revision also summarizes the progress made in the program to date, and the activities remaining to complete the program.

Two HTGR concepts—a pebble bed reactor (PBR) and a prismatic modular reactor (PMR)—are candidates for the NGNP. The designs provide inherent safety, which prevents core damage under all design basis accidents and presently envisioned severe accidents. The principle that guided their design concepts was to passively maintain core temperatures below fission product release thresholds under all accident scenarios. The required level of fuel performance and fission product retention reduces the radioactive source term by many orders of magnitude and allows potential elimination of the need for evacuation and sheltering beyond a small exclusion area. This potential, of safety however, mandates on exceptionally high fuel fabrication quality and performance under normal operation and accident conditions. Germany produced and demonstrated high-quality fuel for their PBRs 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 early development activities to ensure concept viability.

The design specifications for the NGNP are not yet fully defined, but the maximum burnup envisioned in a prismatic HTGR is within the range of 150 to 200 GWd/MTHM (metric tons of heavy metal) or 16.4 to 21.8% fissions per initial metal atom (FIMA).[1] The composition of the fuel kernel and the fuel element geometry have not been decided at this time, with both pebble bed and prismatic designs being considered for NGNP deployment. Although Germany has demonstrated excellent performance of silicon carbide (SiC) tristructural-isotropic (TRISO)-coated UO_2 particle fuel up to about 10% FIMA and 1,150°C, UO_2 fuel is known to have limitations because of CO formation and kernel migration at the high burnups, power densities, temperatures, and temperature gradients that may be encountered in the prismatic NGNP design. With uranium oxycarbide (UCO) fuel, the kernel composition is engineered to prevent CO formation and kernel migration, which are key threats to fuel integrity at higher burnups, temperatures, and temperature gradients. Furthermore, the excellent performance of German SiC TRISO-coated UCO fuel up to 22% FIMA (as measured by the in-pile gas release in irradiation test FRJ-P24[2]) gives added confidence that high-quality SiC TRISO-coated UCO fuel can be made and its performance statistically demonstrated.

In addition to excellent fission product retention during normal operation at high burnups and high temperatures, the NGNP fuel must exhibit satisfactory fission product retention under postulated accident conditions. Limited data on the accident performance of SiC TRISO-coated UO_2 fuel at high burnups indicate increased cesium releases at burnups ≥14% FIMA[3], so safety testing is an important element of the program

An independent review[4] of technology uncertainties and alternatives for meeting the functional objectives of the NGNP recommended that fuel development for the NGNP should initially focus on coated-particle fuel using a UO_2 kernel to minimize schedule risk for successful fuel qualification. The NGNP would be limited with respect to the desired fuel performance requirements, but subsequent development and testing of fuel with a UCO kernel as described in the original program plan[1] (and recommended by the review panel) would offer the possibility of achieving the desired fuel performance. The AGR Fuel Development and Qualification Program chose to continue its initial plan to develop particle fuel using a UCO kernel to minimize the total time and resources required to qualify a fuel to meet NGNP fuel performance requirements. Thus, SiC TRISO-coated UCO was chosen as the baseline fuel to be fabricated and tested in this program. A PMR fuel element design was chosen to be consistent with past fuel fabrication experience in the U.S. This fuel development path complements particle fuel development with a UO_2 kernel that was being pursued by South Africa, China, Japan, and Europe.

The NGNP fuel acquisition strategy study[5] addressed options for supplying NGNP fuel for both prismatic and pebble bed reactor designs. In that report, a detailed technical assessment of potential fuel vendors for the first core of NGNP was conducted by an independent group of international experts based on input from the three major reactor vendor teams. Parts of the assessment included an evaluation of the credibility of each option, along with a cost and schedule to implement each strategy compared with the schedule and throughput needs of the NGNP project. While credible options were identified, many changes in the assumptions underlying the strategy and in externalities that have happened in the interim require that the options be re-evaluated. This re-evaluation is currently under way.

Early efforts of the AGR Fuel Development and Qualification Program have been focused on prismatic fuel because of the broad body of existing data applicable to pebble bed designs and the ongoing international pebble bed fuel development activities. This approach was necessary to maintain

the prismatic concept as a viable option until a design was selected for NGNP. The results of the fuel acquisition strategy study reinforced this perspective and confirmed the need for the activities identified in the AGR fuel program plan, which, in the early program tasks, are largely independent of the reactor design selected. It was concluded that the logic underlying the AGR fuel program plan remains valid. If a PBR design should be developed for the NGNP, UCO fuel could be compacted into spherical elements to fit this design, although variations in particle design and qualification service conditions may be needed for optimal performance. Since international pebble bed UO_2 TRISO fuel programs have been recently curtailed, options for pebble bed fuel qualification need additional study. Future revisions to this plan will address this issue in more detail if the PBR concept is selected or required to remain as a viable option for NGNP.

The TRISO-coated UCO fuel specification[6] utilizing SiC as the primary fission product retention layer and developed in response to the extensive evaluations[7,8] of the fuel failures in the New Production Reactor (NPR) and High Flux Isotope Reactor (HFIR) Removable Beryllium (HRB)-21 irradiations was the starting point for the fuel specification developed for this program.[9] It is expected that this fuel will permit high-burnup operation (16-22% FIMA) at fuel temperatures up to 1,250°C with fast fluences to 5×10^{25} neutrons/m^2, supporting operation in the lower range of projected NGNP core outlet temperatures (750 to 950°C).

This program plan identifies the research and development (R&D) needed in the areas of fuel fabrication manufacture, fuel and materials irradiation, safety testing and post-irradiation examination (PIE), fuel performance modeling, and fission product transport and source term. An updated integrated schedule and budget for the work required to develop, scale-up to a production capability, and transfer a coated-particle fuel fabrication capability to industrial organizations within the U.S. is provided.

In the late 1980s, Germany demonstrated coated-particle fuel performance to the desired level of quality and predictability in the Arbeitsgemeinschaft Versuchsreaktor (AVR) and several materials test reactors. The starting point for the present program was a fuel design based on the most recent gas turbine modular helium reactor fuel product specification, combined with the successful German-like coating process. The basic structure of this program is delineated into the major task areas below:

- Fuel Fabrication
  - Utilize publicly available German coating process information and German fuel and material property data
  - Perform additional characterization of the coating properties of the German particles as needed
  - Use 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 verify that these coating parameters yield German coating properties in the PMR NGNP particle design
  - Reestablish basic quality control (QC) capability for coated-particle fuel and develop new QC methods (as required) for enhanced characterization of kernels and coatings
  - Develop an improved fuel compact fabrication process using a thermosetting resin-based matrix
  - Produce and characterize initial reference fuel particles and selected variants for shakedown irradiation testing
  - Produce 1 inch-long by 0.5 inch-diameter compacts for shakedown irradiation testing

- Assemble a prototypic size coater and set/test the reference coating process parameters with the selected equipment
- Produce updated reference fuel and selected variants with a prototypic size coater for fuel performance testing
- Produce driver and designed-to-fail (DTF) fuel particles for fission product transport testing
- Produce final reference fuel with a prototypic-size coater for fuel qualification testing
- Reduce fuel fabrication risk by developing automated QC and fuel process feedback technologies that can be implemented in a fuel fabrication pilot plant to support cost effective fuel production
- Transfer the fuel fabrication and QC technology to industrial organizations interested in deploying the NGNP.

- Fuels and Materials Irradiations, Safety Testing, and PIE
  - Perform shakedown irradiation, safety testing, and PIE of the initial reference fuel and selected variants from laboratory scale equipment to obtain normal operation and accident condition performance data (AGR-1)
  - Perform irradiation, safety testing, and PIE of both UCO and UO_2 TRISO fuel from prototypic scale equipment to obtain normal operation and accident condition performance data (AGR-2)
  - Perform irradiation, safety testing and PIE of representative fuel containing DTF particles in support of fission product transport model development (AGR-3/4)
  - Perform irradiation testing, safety testing, and PIE of the qualification test fuel to demonstrate that the reference fuel meets NGNP fuel performance requirements for normal operating conditions and accident conditions (AGR-5/6)
  - Perform irradiation, safety testing and PIE of the qualification test fuel in support of fuel performance and fission product transport code validation (AGR-7/8).
- Fuel Performance Modeling
  - Improve the existing coated-particle material property database to support development of constitutive relations that describe the thermomechanical, thermophysical, and physiochemical behavior of coated particles
  - Develop a mechanistic fuel performance model for normal and off-normal NGNP conditions and benchmark against relevant performance data.
- Fission Product Transport
  - Evaluation of data from irradiation and safety testing of DTF fuel to characterize fission product release and transport from TRISO-coated particles under normal and off-normal NGNP conditions
  - Perform out-of-pile helium loop testing to characterize fission product deposition on and re-entrainment from primary system surfaces under normal and off-normal NGNP conditions
  - Improve and benchmark existing fission product transport models.

- Validate the fuel performance and fission product source term models using data from the irradiation and safety testing.
- Conduct experiments to validate the NGNP source term models.

An underlying theme in the work scope is the need to improve understanding of the relationship between the fuel fabrication process, fuel product properties, and in-reactor performance. Fuel performance modeling and analysis of fission product behavior in the primary circuit and reactor building are also addressed. The performance models are considered essential for several reasons, including guidance for the plant designer in establishing the core design and operating limits and demonstration for the licensing authority that the applicant has a thorough understanding of the in-service behavior of the fuel system and extrapolation of test results to the reactor. The fission product behavior task will provide primary source term data needed for design and licensing.

A logical extension of the irradiation and safety testing activity is to establish the operating margins for the fuel. For NGNP fuel, this would mean measuring the fuel performance at 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 accident conditions. The irradiation test train designated AGR-7 will be designed so that some measurable level of fuel failure and/or fission product release is expected to occur in support of fuel performance model validation and operating margins.

As others in the international community continue their fuel development for the very high temperature reactor (VHTR), opportunities for collaboration occur. A VHTR Fuel and Fuel Cycle Project Management Board has been established under the Generation IV International Forum (GIF) VHTR project to identify areas of collaboration. As of the writing of this revision (Revision 3) of this plan, collaborative activities have been identified and are under way. They are mentioned in the relevant sections of the document.

2. GOALS, ASSUMPTIONS, AND REQUIREMENTS

An overall set of programmatic goals, assumptions, and requirements has been developed to guide the preparation of this Technical Program Plan and is presented here. The scope of the plan is subdivided 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 requirements developed to guide the planning of each of these program elements are discussed in Section 3. A high-level set of goals, assumptions, and requirements from the perspective of the overall program are identified in the following sections.

2.1 Overall Program Goals

The overall goals for the program are to:

- Provide a fuel qualification data set in support of the licensing and operation of the NGNP. Gas-cooled reactor fuel performance demonstration and qualification comprise the longest duration research and development task required for NGNP design and licensing. The fuel is to be demonstrated and qualified for service conditions enveloping normal operation and accident conditions.
- Support deployment of the NGNP for hydrogen 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 resources (e.g., GIF VHTR activities).

Fuel qualification is herein defined as the demonstration of the robust performance and efficacy of the reference coated-particle fuel through production of experimental data and analysis results. This fuel qualification effort is meant to support the Nuclear Regulatory Commission (NRC) in its preapplication review for the NGNP concept and to support the NRC in its eventual issuance of a license.

2.2 Overall Program Assumptions

Overall program assumptions include:

- Government and potential industry co-sponsors of the NGNP recognize that a stable, long-term, disciplined fuel development and qualification effort offers the greatest probability of success.
- Coated-particle fuel fission product retention of 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 NGNP designs may impose more demanding service conditions than the German High Temperature Reactor (HTR) Modul and require testing of a fuel based on the NGNP design and the German coating process.

- It is technically feasible to reestablish a production capability equivalent to the German capability in the U.S. at reasonable cost and on a schedule consistent with initial plant deployment schedule(s).
- A base technology program aimed at reestablishing the capability to fabrication and test fuel, with a follow-on goal of improving the technology to the point where it can support economic deployment of HTGR, is the lowest-risk approach to achieving the program goals.
- The target peak time-averaged fuel temperature (1,250°C) can support NGNP operation at least to the lower end of the anticipated design core-exit helium coolant temperature range (750 to 950°C).
- Adequate annual DOE funding will be available to support the fuel development and qualification activities outlined in this Technical Program Plan.
- Results of the AGR fuel fabrication program will be responsive to design data needs of the fuel vendors and to the NRC's fundamental data needs for vendor Appendix B licensing.
- 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, QA and QC are outside the scope of this program.
- Commercialization of fuel manufacturing technology, including qualification/licensing of vendor-supplied fuel, will proceed in parallel with the later phases of this program. Industry investment in fuel fabrication capability will expand commensurate with progress made in the early phases of development and as necessary to support initial plant deployment schedule(s).
- No major programmatic or technical difficulties that could significantly impact the schedule are encountered during the development, irradiation testing, and subsequent experiments.

2.3 Overall Program Requirements

Key requirements for the program are delineated below.

- Establish an AGR TRISO fuel development and qualification program that will:
  - Address the generic issues previously identified by NRC staff members in their preapplication reviews of the prismatic MHTGR and the pebble bed modular reactor concepts
  - Produce a fuel fabrication specifications adequate to fabricate fuel that meets the anticipated performance requirements of the reactor designer
  - Reestablish the domestic capability to fabricate high-quality coated-particle fuel using prior U.S. experience and nonproprietary German coating product characteristics and process data
  - Improve understanding of the fabrication process, its impact on as-fabricated fuel properties, and in turn, their impacts on in-reactor performance
  - Reestablish the domestic capability to manufacture prismatic and/or pebble bed fuel elements (consistent with NGNP design options)
  - Produce or obtain coated-particle fuel samples as needed to support required testing
  - Complete the design and construction of in-reactor test trains for domestic irradiation testing of coated-particle fuel

- Develop and qualify coated-particle fuel through generation and presentation of statistically sufficient irradiation, post-irradiation safety testing, and PIE data under normal operation and accident conditions consistent with anticipated designer requirements
- Demonstrate sufficient margin to failure for this fuel form under normal operating and accident conditions
- Improve understanding of fuel behavior and fission product transport, and use that understanding to improve the fuel performance and fission product transport models to predict fuel behavior and fission product transport under normal operating, operational transient, and accident conditions to the required predictive accuracies
- Develop pertinent technical information that supports the selection and implementation of the fabrication process used by the NGNP fuel vendor
- Develop pertinent technical information that can supplement the NGNP fuel vendor's own licensing/qualification data in the topical report supporting NGNP licensing.

- Implement this plan such that it supports both prismatic and pebble-type fuel designs. The effort dedicated to each design will be proportionate to the associated level of industry interest and commitment and the level of fuel development and qualification activities under way around the world. Early stages of the program should support both designs by concentrating on TRISO fuel particle performance, which is common to both designs.
- Focus fuel fabrication process development on low enriched uranium (LEU) UCO-SiC TRISO fuel for a prismatic-type reactor. Document, as appropriate, the applicability of fuel fabrication process development to pebble-type fuel production.
- Implement this plan in accordance with the DOE QA requirements specified in 10 CFR 830, "Nuclear Safety Management," Subpart A, "Quality Assurance Requirements" and in DOE Order 414.1C, "Quality Assurance." 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)-1-2000, "Quality Assurance Requirements for Nuclear Facility Applications," 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 elements as appropriate.

3. PROGRAM ELEMENTS

This section summarizes the more detailed goals and requirements associated with the individual program elements and the activities required to meet these and the high-level goals and requirements identified in Section 2. Program elements include fuel fabrication, fuel and materials irradiation, safety testing and PIE, fuel performance modeling, and fission product transport and source term.

3.1 Fuel Fabrication

3.1.1 Goals, Assumptions and Requirements

The goals, assumptions, and requirements specific to this program element are as follows detailed below.

Goals

1. Establish coated-particle fuel fabrication technology in the U.S. for the NGNP that is capable of producing fuel at a quality level at least as good as produced by German fuel technology. Establish this capability on a schedule consistent with the NGNP deployment schedule.
2. Develop fundamental understanding of the relationships between fuel fabrication process parameters, fuel product properties, and fuel normal operation and accident condition performance.
3. Develop automated fuel fabrication technology suitable for mass production of coated-particle fuel at an acceptable cost and at a high quality and consistency level.
4. Establish fuel process and product specifications that define the requirements the fuel must meet to ensure acceptable performance within the envelope of the NGNP normal operation and accident fuel service conditions.
5. Develop and document the manufacturing processes required to meet the fuel process and product specifications that will be developed to satisfy Goals 3 and 4.

Assumptions

1. The coated-particle design to be qualified in the AGR program will be based on the most stringent performance requirements for different types of HTGRs (pebble bed and prismatic). This approach will result in the qualification of a fuel performance envelope that can then be used by either HTGR technology.
2. Fuel that is capable of acceptable normal operation and accident condition performance up to a target peak time averaged fuel temperature of 1,250°C in normal operation can support NGNP operation within a substantial portion of the anticipated core-outlet helium coolant temperature range (750 to 950°C).
3. The capability to mass-produce high-quality, coated-particle fuel pebbles or prismatic blocks economically is a prerequisite for commercial viability of prismatic HTGRs.
4. The LEU UO_2 particles qualified by the Germans for burnup up to about 10% FIMA are not adequate for higher fuel burnup (16–22% FIMA) and higher temperature and temperature gradient service in HTGRs.
5. Fuel particles made with UCO kernels and having coating properties equivalent to those of German fuel particles from composite EUO 2358-2365, which were irradiated in the HTR Modul proof tests

(High Flux Reactor [HFR]-K5 and HFR-K6) with no in-pile failures, will perform well in fuel compacts under HTGR irradiation conditions.

6. The lowest risk path to successful coated-particle manufacturing is to replicate the proven German coating technology to the extent possible on a coated particle design, which incorporates the lessons learned from prior U.S. fabrication and irradiation experience.

Requirements

1. Reestablish and demonstrate coated-particle fuel fabrication capability, from kernel production through fuel element production.
2. Conduct fuel kernel process studies to optimize the UCO kernel fabrication process (e.g., carbon dispersion, broth chemistry, calcination, carburization, and sintering).
3. Conduct fuel coating process studies to determine the adaptability of the German-like coating process to NGNP fuel and to establish coating conditions that yield coating layers having microstructural properties and features comparable to the coating layers on the German fuel particles in proof-test composite EUO 2358-2365.
4. Replicate the German coating technology with a process suitable for large-scale fuel production. This will be accomplished by using a coater that provides a coating environment similar to the coating environment in the German production coater and that has appropriate features for a production coater (for loading, unloading, sampling material from the coater, and cleaning).
5. Develop improved QC methods, as required, to improve the characterization of the fuel and to support large-scale fabrication of NGNP fuel.
6. Fabricate fuel as needed for irradiation testing, including DTF fuel for fission product transport tests. The test fuel shall meet the product requirements specified in the test fuel product specifications prepared for the individual irradiation tests. These fuel product specifications will be based on the specific requirements for each irradiation test.
7. Prepare a fuel product specification and process specification for large-scale NGNP fuel fabrication that defines all requirements that the fuel must satisfy to ensure acceptable fuel performance under NGNP operating and accident conditions.
8. Develop automation technologies that can be applied to fuel fabrication processes.

3.1.2 Scope of Fuel Fabrication

The ultimate fabrication goal for HTGR fuel is the economical production of high-quality kernels, coated particles, and compacts or pebbles that meet the fuel product specifications. The fuel fabrication activities described here are intended to develop and qualify a fuel fabrication process that can serve as the foundation for fabrication commercial-scale, coated-particle fuel for advanced gas-cooled reactors. These activities must optimize the process to achieve the required kernel, coated-particle, and compact or pebble characteristics and quality, and scale-up of kernel production, coating, and compact or pebble fabrication processes.

Coated-particle fuel fabrication is considerably different from light water reactor (LWR) fuel manufacturing. The fabrication process developed within the AGR program begins with 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 then dried and sintered into hard ceramic particles yielding kernels of a controlled and consistent size and chemistry.

Fuel kernels are coated using fluidized-bed chemical vapor deposition processes. The coatings include a low-density carbon layer (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 accident conditions. The finished coated particle is a small (~1 mm OD) carbon and ceramic sphere that is stable to temperatures well beyond 1,600°C.

The coated fuel particles are formed and processed into physical shapes for use in the reactor (see Figure 1). For the prismatic fuel 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. For the pebble bed design, fuel particles are pressed into tennis ball-size spheres that are recirculated in the reactor. For both designs, the particles are over-coated with a carbonaceous matrix composed of graphite powder and a thermosetting resin binder, formed into the desired shape, then carbonized and high-temperature heat treated to provide a thermally stable material.

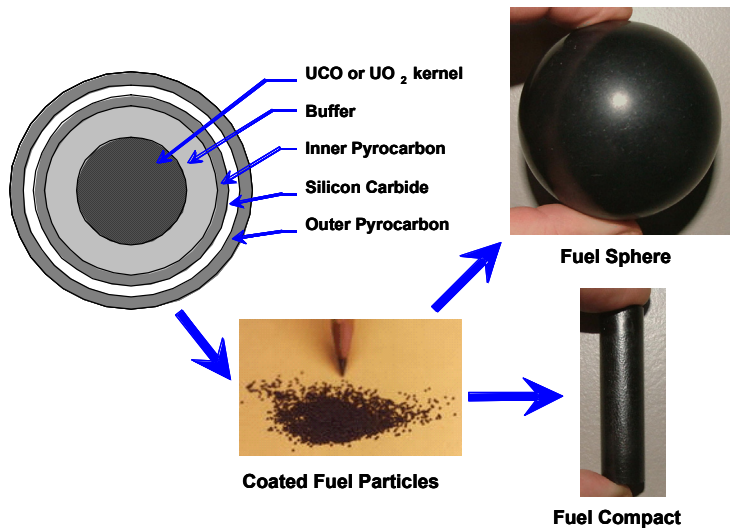

Figure 1. Potential fuel forms.

The target quality level for 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-Modul proof tests[10] taken as a standard for comparison, in combination with core design driven quality specifications derived during the gas turbine modular helium reactor (GT-MHR) conceptual design.[7] The AGR program fuel fabrication effort was designed to expand understanding of the relationship between kernel and coating properties, fabrication process conditions, and the irradiation performance of the fuel. The earlier U.S. and German manufacturing efforts and the 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[7, 8] along with U.S. and German fuel fabrication processes and irradiation performance.[11] These studies suggest key differences between German and historical U.S. coating processes and coating properties that contribute to better irradiation performance. The most significant differences in the German processes are (1) greater deposition rate of pyrocarbon layers resulting in more isotropic coatings having greater stability under irradiation to high fast neutron fluence; (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, possibly, in beneficial effects on coating properties; and (4) lower SiC coating temperature, resulting in smaller grain size. Thus, the starting point for fuel fabrication development was the U.S. kernel and compacting

experience coupled with knowledge about the German coating process and coating properties and supplemented by the lessons learned from fuel technology development within the U.S.

The work to produce coated-particle fuel that meets the specifications includes kernel process development, coating process development, compacting process development, QC methods development, and process documentation. The plan also includes work to develop automated fuel fabrication technology suitable for mass production of coated-particle fuel at an acceptable cost. The scope of fuel manufacture activities is summarized below, with a discussion of the accomplishments and status in as of the date of this revision Section 5.1.

3.1.2.1 Prepare Irradiation Test Fuel Specifications

Developing a fuel fabrication process and fabricating irradiation test fuel in a manner that complies with the quality assurance requirements of NQA-1[12] is based on the specification of kernel, coated particle and compact properties and on key process conditions. Detailed product specifications, along with a limited set of process conditions affecting microstructure characteristics that are known to be important to irradiation performance but cannot be fully characterized, are required for each irradiation to be conducted within the program. These specifications include the parameters identified in Table 1. (Property specifications include both properties for individual batches as well as for composites formed from multiple batches.)

Table 1. Fuel specification parameters.

Parameter	Mean	Critical Region[a]	Fraction in Critical Region
Kernel Batch			
Envelope Density	X		
Diameter	X		
C/U ratio	X		
Sphericity		X	X
Microstructure		Measurement Only	
Kernel Composite			
^{235}U Enrichment	X		
C/U ratio	X	X	X
O/U ratio	X		
(C+O)/U ratio	X		
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
Total U	X		
Sphericity		X	X
Coated Particle Batch			
Thickness (Buffer, IPyC, SiC, OPyC)	X		

Parameter	Mean	Critical Region[a]	Fraction in Critical Region
Missing OPyC	X		
Coated Particle Composite			
Thickness (Buffer, IPyC, SiC, OPyC)	X	X	X
Density (Buffer)	X		
Density (IPyC, SiC, OPyC)	X	X	X
Anisotropy (IPyC, OPyC)	X	X	X
SiC Aspect Ratio		X	X
Defective SiC Fraction	X		
Missing OPyC	X		
SiC Inclusions[b]	X		
SiC Microstructure		Visual Standards	
Heat Treated Compacts			
Uranium Loading	X		
Diameter	X	X	X
Length	X	X	X
Matrix Density	X		
Impurity Content (Fe, Cr, Mn, Co, Ni, Ca, Al, Ti, V, Cl)	X		
Heavy Metal Contamination	X		
Defective SiC Fraction	X		
Defective IPyC Fraction	X		
Defective OPyC Fraction	X		
Coating Process			
Coating Gas Temperature (Buffer, IPyC, SiC, OPyC)	X		
Coating Gas Fraction (Buffer, IPyC, SiC, OPyC)	X		
Coating Gas Ratio (IPyC, OPyC)	X		
Coating Rate		Measurement Only	
Compacting Process			
Molding Pressure	X		
Carbonization Heating Rate, Temperature and Hold Time	X		
Final Heat Treatment Heating Rate and Temperature	X		

a. The specification of a critical region boundary as well as the fraction of particles within the critical region is provided to limit the distribution tail of a property.

b. An indication of defects within the SiC layer, historically identified by General Atomics (GA) as "Gold Spots," but not detectible as such in the more opaque finer SiC microstructure more typical of German fuel.

Executing this plan develops specifications for the fuel produced for the series of irradiation tests, leading to a specification for fuel to be produced for the reactor.

3.1.2.2 Fuel Kernel Manufacturing

As discussed earlier, a UCO kernel was selected for developing coated-particle fuel to support the NGNP project. UCO kernels had been produced for earlier irradiation testing in the U.S. by General Atomics (GA),[13,14] and Babcock & Wilcox (B&W, formerly BWX Technologies or BWXT).[15] Today the B&W fuel facility is the only facility that can handle enrichment levels in excess of 5%. The B&W internal gelation UCO kernel production process was selected for the AGR fuel development 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 for the AGR fuel program. Thus, the scope of fuel kernel manufacturing includes the following elements:

- Process development:
  - Achieve specified kernel density
  - Improve carbon dispersion in the acid-deficient uranyl nitrate (ADUN) solution used in kernel formation
  - Optimize the sintering process
  - Reduce process variability
- Produce natural uranium oxycarbide (NUCO) kernels to support coating process development
- Produce low enriched uranium oxycarbide (LEUCO) 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 were within the U.S. The GA coater used to coat the fuel irradiated in HRB-21, NPR capsules, had been shut down for over a decade, and the facility had been completely dismantled. Small coaters remained at Oak Ridge National Laboratory (ORNL) and B&W, but neither had been operational for the production of TRISO fuel for many years. Thus, U.S. TRISO particle-coating capability needed to be reestablished. In addition, root cause assessments of the HRB-21 and NPR irradiation particle failures[7, 8, 11] indicated a need to adjust the coating process parameters to change the high-density pyrocarbon and silicon carbide layer microstructures. Given the successful performance of pyrocarbon and silicon carbide coatings produced by the German program, a primary objective is the identification of process parameters that would produce coating characteristics equivalent to German coatings. Another objective is the development of 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 are required during initial coating development to obtain process conditions and durations that produce the desired coating properties and thicknesses. These runs are conducted in a laboratory-scale coater to limit the cost and quantity of materials required. The assessments noted above concluded that a focus on limiting uranium dispersion during application of the silicon carbide 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 includes a study of the relationship between IPyC coating conditions, IPyC layer permeability, and IPyC properties that influence irradiation performance (density, anisotropy, and surface connected porosity).

Laboratory-scale coater runs establish the process conditions needed to produce particles that meet the specifications and improve understanding of the relationship between process parameters and key coating properties, but significant uncertainties regarding the relationship between properties and irradiation performance remain. A reference fuel specification plus variations in key coating parameters is, therefore, needed to provide confidence in achieving acceptable performance in the first irradiation. Producing multiple coated-particle types while meeting the AGR fuel program schedule within funding constraints required the fuel for the first irradiation and initial fission product transport irradiations to be produced in the laboratory-scale coater.

Production of the quantities of fuel required to support initial NGNP operation will require a larger coater, and fuel qualification must be based on fuel produced in the larger coater. Thus, coater scale-up issues need to be addressed in the context of defining the coater size and configuration for production of the particles used in subsequent irradiations and the production of fuel for the NGNP. The results of the small coater operation are used to reduce the number of large coater runs needed to achieve the specified coating properties, but a significant process development scope is still needed to define the large coater process conditions. The large coater is then used to produce the coated particles needed for the remaining irradiations.

3.1.2.4 Compacting Process Development

Historically, the 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 temperature. The alumina powder used in the carbonization process was another source of impurities that can potentially attack the SiC coating.

A thermosetting-resin based matrix process is preferred for large-scale fuel manufacturing for two reasons. First, it should result in improved fuel quality because (1) the thermosetting matrix is formulated from raw materials having lower levels of impurities; (2) the thermosetting matrix yields stronger, less-friable compacts; and (3) the thermosetting matrix process involves lower compacting forces and less handling of the compacts, thereby reducing the potential for damage. Second, it eliminates the need to pack the compacts in alumina powder during carbonization, and therefore, it is better suited to automation, which would reduce the cost of fuel compact fabrication.

Based on these considerations, the AGR program is developing the thermosetting-resin process using a matrix formulation equivalent to the matrix used in the German proof test fuel spheres. Development scope includes:

- Replicating the matrix formulation
- Establishing the equipment and process conditions needed to uniformly overcoat particles with the matrix
- Establishing the equipment and process conditions needed to form the over-coated particles into compacts and perform carbonization and final heat treatment
- Producing compacts needed for characterization and irradiation in the AGR program.

3.1.2.5 QC Methods Development and Application

QC methods are needed to demonstrate that the fuel fabricated for the AGR fuel program complies with the product specifications. As with the coating process, facilities were unavailable to measure the properties identified in EDF-4380, "AGR-1 Fuel Product Specification and Characterization Guidance,"[16] at the required confidence levels (typically 95% confidence). The development of QC methods therefore involves reestablishing traditional characterization procedures at both ORNL and B&W, and developing advanced new methods, mainly at ORNL. The following capabilities are needed for the inspection and testing of kernels, coated particles, and compacts to demonstrate compliance with fuel product specifications:

- Chemistry of kernel batches and composites (carbon, oxygen, uranium and 15 impurities), including kernel-to-kernel variability in the UCO kernel stoichiometry
- Kernel U-235 enrichment
- Ceramography to provide images for coated-particle analysis
- Automated image analysis for kernel and particle diameter, sphericity, and coating thickness measurements
- Density gradient column for PyC and SiC sink-float density measurements
- Mercury porosimetry for measuring kernel, buffer and coated-particle envelope density and for PyC surface connected porosity measurements
- A technique for measuring PyC coating anisotropy
- Compact measurements including length, diameter, mean uranium loading, total mass, and defective IPyC and OPyC coating fractions
- Leach-burn-leach testing of fuel compacts to determine the defective SiC and uranium contamination fractions, and the quantity of specified impurities outside the SiC layer
- Visual examination of burned-back particles to identify SiC inclusions.

3.1.2.6 Fuel Product and Process Documentation

The description of fuel fabrication development and irradiation and safety testing in this plan, when combined with additional reactor design information, provides the necessary information to finalize the top-tier fuel product specifications that define the requirements for fuel to be used in the NGNP. Additional reports will be produced to document process and QC development, and 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 successfully fabricate fuel that consistently meets the fuel product specifications and the performance requirements of the NGNP, and the allowable process variations (to the extent determined by the process development tasks).

3.1.2.7 Technology to Mass-Produce Particle Fuel

Demonstrating economic feasibility is crucial to deployment of the NGNP, and a significant component of NGNP costs will be for constructing and operating the fuel fabrication facility. While obtaining cost estimates for such a facility is normally part of the facility design effort, an understanding of fuel fabrication costs is needed to identify development activities aimed at reducing high cost fuel fabrication steps and operations, and to ensure that processes used in making test fuel are economically

viable. While scale up of any manufacturing facility to production throughputs via procurement of multiple equipment items and lines demonstrated at small scale represents a low-risk pathway in terms of technical risk, this approach usually results in the highest cost compared to alternative approaches.

Based on a preliminary 600 MW(t) prismatic NGNP design,[17] a first core load containing 5,600 kg U in 425-µm-diameter kernels and compacts with a 35% packing fraction will require production of approximately 5 million 1 inch-long compacts, 18 billion (17,600 kg) coated particles, and 20 billion (8,640 kg) kernels, assuming 90% efficiencies at each stage of fuel fabrication. For comparison, for the AGR-2 experiment (UCO fuel portion only), a total of 7.5 kg of kernels , approximately 17 kg of particles and 1,009 compacts were produced as part as the overall fabrication campaign.

To reduce the entry-level risk for an industrial sponsor of a HTGR, it is not only necessary to qualify the fuel, but also to show that the fuel can be fabricated economically. The following tasks included in this work area will generate the needed data and technology to support future efforts to achieve a cost-competitive HTGR fuel manufacturing capability:

- Evaluate options for producing kernels, coated particles, and compacts at costs[c] and rates required for a full-scale fuel fabricated facility.
- Select approaches from the above evaluations, and develop and demonstrate equipment and processes capable of achieving the required costs and production rates.
- Develop automated QC methods that are adequate to support large-scale production. These methods should be nondestructive and capable of high throughput rates. Develop advanced QC methods to reduce loss of fuel with destructive and nondestructive evaluation methods. Establish feasible methods to enable 100% inspection and to provide near real-time feedback to the fuel fabrication processes.
- Develop a conceptual design for an automated fuel fabrication plant.
- Prepare a waste minimization plan for future implementation in a fuel fabrication plant.
- Perform a cost evaluation to develop estimates of the fuel unit cost for a commercially viable HTGR fuel fabrication plant and the fuel unit cost potentially achievable in a fuel fabrication plant utilizing the reference manufacturing processes. These cost estimates are needed to provide guidance to the pilot plant design efforts and identify process modifications that may be necessary to reduce fabrication costs to a more acceptable level.

An initial estimate of the fuel cost from a full-scale production facility is needed to provide near-term guidance to fuel fabrication development activities, ensure processes used to produce test fuel provide adequate data to support the design of the full-scale facility, and to support NGNP cost studies and evaluations. The fuel cost estimate will be broken down into components such as feedstock, utilities, labor for each fabrication and upgrading process, QA/QC analysis labor and materials, etc., in order to identify the most critical areas needing development to improve fuel fabrication economics.

The initial fuel costs will be based on several assumptions about the NGNP size and fuel design, scaled process designs, waste treatment, sampling, and quality analyses. Process flow diagrams and

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c. Cost considerations include generation and disposal of hazardous waste, and accommodation of fire and/or explosion hazards in the design and operation of the processes as well as the direct costs associated with fuel fabrication.

material balances for full-scale facilities will be developed and used as the basis for sizing major equipment and estimating operating costs.

A more thorough and accurate estimate of fuel costs should be performed in a second evaluation as part of the conceptual design for an automated fuel fabrication plant and following:

1. More definitive requirements for the NGNP fuel (fuel type, production rate, kernel diameter, kernel enrichment(s) etc.)
2. Studies improving efficiencies or capacities of existing processes
3. Studies to define waste treatment steps
4. Preparation of a fuel characterization plan.

The latter three of these activities plus preparation of a conceptual design are described below.

Process Improvement Studies and Evaluations

Studies or evaluations may be defined based on uncertainties in the assumptions used in the initial fuel cost estimate or recommendations from the conceptual design (see below). These studies could, for example, evaluate larger diameter kernel fabrication equipment, evaluate ways to reduce waste and improve yields, operate the forming column in a continuous mode, and eliminate manual transfers.

Waste Treatment Evaluation and Waste Minimization Plan

The process flow diagrams and material balances developed for the initial fuel cost estimate will serve as the starting point for waste treatment evaluations and developing a waste minimization plan. Waste and scrap from AGR-1 and AGR-2 fuel fabrication processes have been stored or disposed in ways that will not be possible for waste from full-scale fabrication facilities. Process changes may be required to avoid mixed (hazardous and radioactive) waste. For example, the kernel forming currently uses trichloroethylene (TCE) as a heat transfer fluid to initiate the chemical reactions within the broth droplets to form green kernels. Over time, the TCE becomes contaminated with uranium compounds, water, acid and other chemical species present in the forming “broth,” and periodic replacement of TCE is required to ensure kernel quality. As TCE is a Resource Conservation and Recovery Act (RCRA) hazardous waste, disposal of uranium-contaminated TCE waste will be problematic. Alternatives to avoid this waste need to be defined and evaluated, such as replacing TCE with a nonhazardous fluid or recovery of the TCE through distillation. The waste minimization plan will consider all scrap and waste streams generated by the fabrication processes and recommend ways to avoid and reduce wastes by process changes, identify streams that can be recycled, discuss unit operations required for recycle, and define waste treatment processes required prior to offsite disposal.

Fuel Characterization Plan and Automated QC Method Evaluations

Characterization of AGR-1 and AGR-2 fuel to determine compliance with specifications required much more calendar time than fabricating the fuels, and it consumed a large fraction of the fuel produced. Ideally, QC methods that are nondestructive, allow for near real-time feedback, allow for large throughputs or 100% inspection are desired for a fabrication facility. Many of the current methods, such as those determining particle defects, are destructive, slow, and require preparation of mounts and manual inspection of images of tens of thousands of particles. As a first step in bridging the chasm between current fuel characterization and practices that contribute to an economical full-scale fuel fabrication plant, a plan will be developed that identifies sample points, sample quantities, sampling frequency, and current characterization methods. The plan will provide an initial evaluation of whether new techniques

are needed, whether automation of current techniques is possible, and if so what development is recommended.

Fuel Manufacturing Plant Conceptual Design

A conceptual design is needed to document the basis for a full-scale fuel manufacturing plant, incorporating results of development and providing a solid basis for fuel costs. The conceptual design along with studies and data, which support the design, provide the source documents to develop risk management plans, safety and hazards analyses, and cost estimates. The basis for the conceptual design starts with input from the NGNP design. A Technical and Functional Requirements document is developed and expanded in a Conceptual Design Criteria document that provides the basis for the conceptual design.

The fuel manufacturing plant conceptual design would include the following:

- A list of requirements documents
- A list of supporting documents
- Process flow diagrams
- Material balances including feedstock, waste and product summary
- Process description
- Studies documenting basis for design decisions and design options
- An equipment list
- Facility description and layout drawings showing equipment arrangements
- Piping and Instrumentation diagrams
- Preliminary electrical, high efficiency particulate air and heating, ventilating and air conditioning drawings
- Environmental safety and health requirements
- Discussion of nuclear safety
- Discussion of safeguards and security
- Discussion of quality assurance
- Staffing requirements and operating cost estimates
- Capital cost estimates
- Preliminary risk assessment
- Schedule.

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 Idaho National Laboratory (INL) and

ORNL. ORNL irradiated fuel for the MHTGR,[18] and both laboratories were involved in irradiation testing of NPR-MHTGR fuel in the early 1990s. The Advanced Test Reactor (ATR) at INL and the HFIR at ORNL are both capable of irradiation testing of AGR fuels, although the HFIR cannot accommodate testing of full-size (60 mm) pebbles. The ATR was selected in large part due to 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. ORNL personnel with irradiation experience participate in technical reviews of the AGR irradiation test trains in the ATR.

3.2.1 Goals, Assumptions and Requirements

The goals, assumptions, and requirements of fuel and materials irradiation, established at the beginning of the program, are as follows:

Goals

- Provide data for fuel performance under irradiation as necessary to support fuel process development, qualify fuel for normal operation conditions, and support development and validation of fuel performance and fission product transport models and codes
- Provide irradiated fuel and fuel materials as necessary for PIE and safety testing.

Assumptions

- Accelerated irradiation (up to 3 times real time in terms of both power and fast flux) is equivalent to or 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.
- Fuel fabrication capability is established to provide fuel samples representative of high-volume production for qualification testing.

Requirements

- Establish the range of irradiation conditions (power, burnup, flux, fluence, temperature, and environment), based on the needs of the reactor designs, to qualify fuel for normal operation
- Establish allowed tolerances on control of irradiation conditions
- Complete the design and construction of test trains for irradiation testing of coated-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 allowed 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 integrated metallic fission product release, such as silver, cesium, and strontium, from fuel in each capsule during irradiation

- Sufficient data to qualify the fuel for normal operation over the required range of irradiation conditions and support code and model development and validation
- Irradiated fuel and material specimens as required to support PIE, post-irradiation phenomenological testing, and safety testing activities.

The selected test train concept to be used in the first two irradiations (AGR-1 and AGR-2) has been placed in Large B positions of the ATR. These test trains contain six capsules that can be independently controlled for temperature and separately monitored for fission product gas release, with each capsule containing twelve 1 inch-long compacts. To increase the capacity for irradiation of fuel and decrease the duration of the irradiations, a test train designed for an ATR flux trap location is under consideration for several of the subsequent irradiations.

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 “B” position red arrow on the right side of Figure 2 points to one of the eight Small B positions, which are adjacent to the driver fuel. The four Large B positions (blue arrow on the right points to the Large B position in the east quadrant of the core), located further from the driver fuel in the beryllium reflector, 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 very well matched to anticipated NGNP conditions. The physics calculations are refined when the actual fuel loadings are known for each test train. The B position irradiations planned in the AGR program are anticipated to take about 2.5 years. Another potential irradiation location is the northeast flux trap (blue arrow at top upper right side), which can accommodate larger test trains with shorter irradiation times. Preliminary calculations indicate irradiation times on the order of 1.5 years in the flux trap location.

For test trains irradiated in a Large B position, continuous gas monitoring capability is made available for each capsule within a test train by providing a set of six dedicated fission product monitors plus an on-line operating spare.

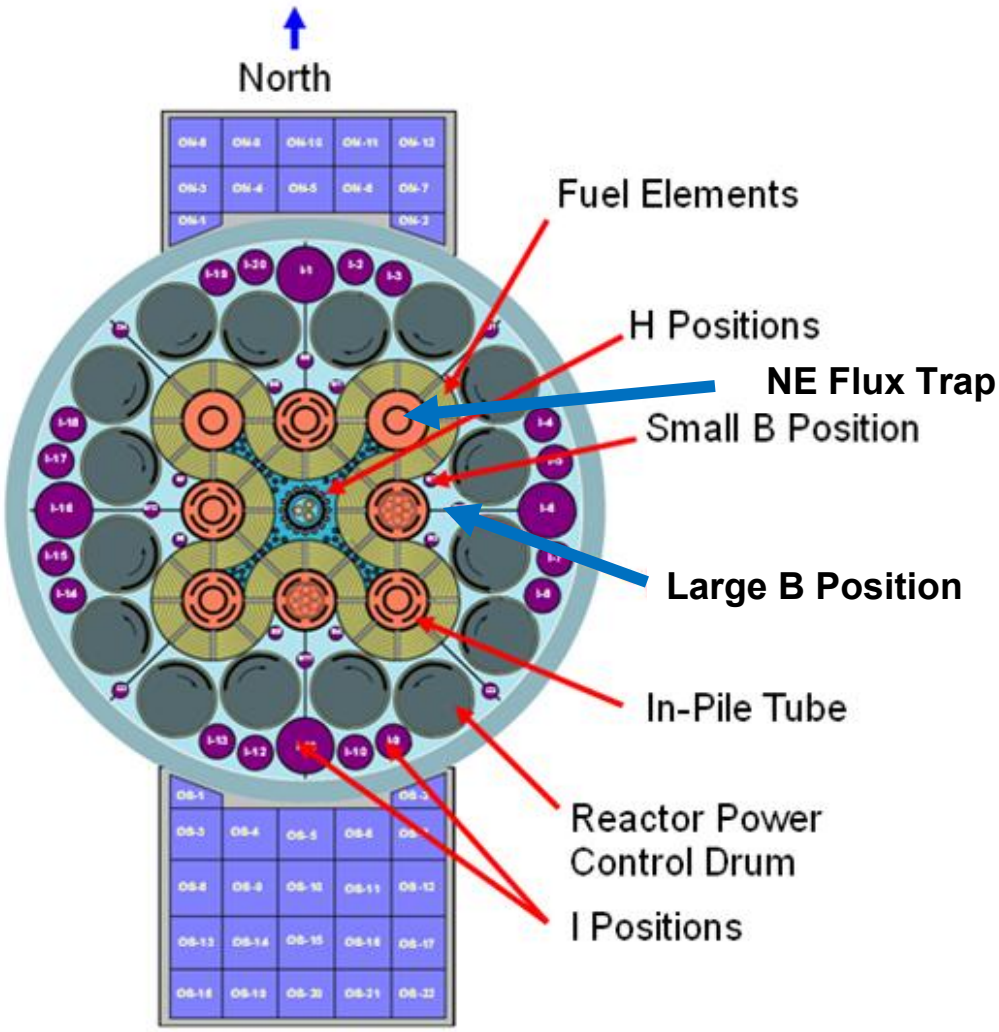

Figure 2. ATR cross section.

Preliminary test train concepts for conducting irradiations in the northeast flux trap location have been developed and analyzed. Results indicate that some of the individual irradiations planned for the Large B positions could be combined for irradiations in the flux trap (e.g., AGR-3 and 4, AGR-5 and 6). For the test trains irradiated in the northeast flux trap, a second set of 12 dedicated fission product monitors plus two on-line operating spares are being installed to provide individual monitoring of up to twelve capsules in a test train with the same capacity of spare monitors used in the Large B position irradiations. The larger diameter of the northeast flux trap location provides greater flexibility for test train design, significantly enhancing the capability for the combined irradiations.

3.2.2 Scope of Fuel and Materials Irradiation

In producing the original version of this plan, the fuel irradiation working group developed a description of the tasks associated with irradiation testing of a representative test train in the ATR and in HFIR. The following tasks were identified:

1. *Test specification*. This task will specify the test articles, irradiation conditions, and results needed to support fuel fabrication and model development and the plant design and licensing. The test specification document will include 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 test.
2. *Test train and supporting systems technical and functional requirements*. This task will establish the detailed requirements necessary to proceed with test train and supporting systems design in accordance with the test 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 quality assurance. It also will include the requirements placed on the experiment by ATR necessary to meet the 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/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 will include procurement or fabrication of test train components in accordance with the specifications, installation, or refurbishment of supporting systems as necessary, and assembly of the test train, including the test articles, ready for insertion into the reactor.
5. *Approval of test articles*. This task includes the receipt, inspection, and acceptance of all test articles (compacts, pebbles, loose particles, 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 insertion into and removal from the reactor, operation of the support and data acquisition systems, documentation of the conditions and results of the irradiation (including establishing a near real-time remote data acquisition), and placement of the test train in its storage location for cooldown.
8. *Cooldown*. This task addresses storage of the test train until the decay heat and radiation levels are sufficiently low to proceed with test train disassembly and shipping (addressed in the PIE scope).

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 test specification, the basic tasks are expected to 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 the irradiation testing.

3.2.3 Planned Irradiations

The number and type of test trains to be irradiated depend on the needs of the fuel manufacture, fuel performance modeling, and fission product transport activities.

The following irradiations were identified based on discussions among the working groups during the course of developing the original plan. Further development of the test train designs may alter the type of test trains to be used for individual irradiations. For example, it may be advisable to conduct the fuel qualification testing with a higher acceleration factor to shorten the schedule if more detailed information on irradiation conditions shows this method to be acceptable.

3.2.3.1 Shakedown/Early Fuel (AGR-1)

This multi-monitored test train included compacts made from particles produced in a small coater in conjunction with fuel process development. This irradiation provided experience with a multi-monitored test train design, fabrication, and operation, and will reduce the chances of test train or capsule failures in subsequent test trains. Having been successfully taken to estimated design burnup and fast fluence, it will provide data on irradiated fuel performance for baseline and fuel variants selected based on data from fuel process development and existing irradiation experience. These early data on performance of fuel variants will support the selection of a reference fuel and development of an improved fundamental understanding of the relationship between the fuel fabrication process, as-fabricated fuel properties, and normal operation and accident condition performance.

3.2.3.2 Performance Test Fuel (AGR-2)

This multi-monitored test train includes compacts containing UCO particles made in a larger coater using process conditions derived from the production of AGR-1 Variant 3 (SiC coatings produced using a mixture of hydrogen and argon diluent gases) to produce coated particles designed for prismatic reactor service. The UCO compacts will be subjected to a range of burnups and temperatures exceeding anticipated prismatic reactor service conditions in three capsules. The test train will also include compacts containing UO_2 particles produced independently by NGNP program participants (B&W, Westinghouse/ Pebble Bed Modular Reactor [Pty] Limited [PBMR] and AREVA) in three separate capsules. The range of burnups and temperatures in these capsules will exceed anticipated pebble bed reactor service conditions. This test train will provide irradiated fuel performance data and irradiated fuel samples for safety testing and PIE for key fuel product and process variants to broaden options and increase the prospects for meeting fuel performance requirements and to support the development of a fundamental understanding of the relationship between the fuel fabrication process, as-fabricated fuel properties, and normal operation and accident condition performance.

3.2.3.3 Fission Product Transport (AGR-3/4)

This multi-monitored test train is a combination of AGR-3 and 4, which were initially planned for separate irradiations in the Large B positions, as shown in Figure 2, but will be combined and placed in the northeast flux trap position in the ATR, also shown in Figure 2. DTF fuel particles for use in fission product transport testing will consist of reference kernels with 10 to 15-µm-thick pyrocarbon seal coatings. These coatings will fail early in the irradiation and provide a known source of fission products. This multi-monitored test train will include compacts of unaltered “driver” fuel particles seeded with DTF fuel particles. The sweep gas will not only contain a mixture of helium and neon necessary to provide thermal control of the experiment but also gaseous impurities (e.g., CO, H_2O) typically found in the primary circuit helium of HTGRs. This will allow an assessment of the effect of impurities on intact and DTF fuel performance and subsequent fission product transport. The test train will be designed to provide data on fission product diffusivities in fuel kernels and sorptivities and diffusivities in compact matrix and graphite materials for use in upgrading fission product transport models. AGR-3/4 will also provide 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 the development of improved fission product transport models.

3.2.3.4 Fuel Qualification (AGR-5/6)

This multi-monitored test train is a combination of AGR-5 and 6, which were initially planned for separate irradiations in the Large B positions, as shown in Figure 2, but will be combined and placed in the northeast flux trap position in the ATR, also shown in Figure 2. The test train will include 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[d] from AGR-1 and 2. This will be the reference fuel design selected for qualification. Variations in capsule conditions (burnup, fast fluence, and temperatures) will be established in accordance with the test specification. The sweep gas will contain helium, neon and a representative set of impurity gases found in the primary system helium of HTGRs. This will allow an assessment of the effect of impurities on fuel performance. One capsule in the test train will contain fuel compacts with DTF particles to support post-irradiation moisture ingress testing. (See R&D Task 3.5.14 and PIE task 21). This test train will provide irradiated fuel performance data and irradiated fuel samples for safety testing and PIE in sufficient quantity to demonstrate compliance with statistical performance requirements under normal operation and accident conditions.

3.2.3.5 Fuel Performance Model Validation (AGR-7)

This multi-monitored test train will include the same fuel type as used in AGR-5/6. The irradiation would test fuel beyond its operating envelope so that some measurable level of fuel failure would occur (i.e., margin test). The test would provide fuel performance data and irradiated fuel samples for PIE and post-irradiation heating test and PIE in sufficient quantity to validate the fuel performance codes and models and to demonstrate capability of fuel to withstand conditions beyond AGR-5/6 in support of plant design and licensing. The sweep gas will be the same as that used in AGR-5/6.

3.2.3.6 Fission Product Transport (AGR-8)

This multi-monitored test train will include compacts seeded with fuel particles that are missing buffers and have different temperatures among capsules. The test train will provide irradiated fuel performance data and irradiated fuel samples for safety testing and PIE to validate fission product transport codes. The sweep gas will be the same as that used in AGR-5/6.

AGR-7 and AGR-8 will be conducted at the same time in the ATR northeast flux trap either housed in two separate test trains or in one test train where half is dedicated to AGR-7 and half to AGR-8.

3.3 PIE and Safety Testing

This program element measures the performance of the AGR fuel under accident conditions and to characterize the state of AGR fuel after irradiation and after accident conditions. PIE and safety testing are strongly interwoven since many of the PIE procedures applied to fuel samples following irradiation are also applied to fuel following safety testing to determine performance in both normal operation and accident conditions. This work will support the fuel manufacturing effort by providing feedback on the performance of kernels, coatings, and compacts. Data from PIE and safety testing, in conjunction with the

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d. The decision to proceed with fabrication of qualification test fuel will be made as early as possible based on information available at the time, which may include 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.

in-reactor measurements (primarily fission gas rate release rate-to-birth rate [R/B]) are necessary to demonstrate that the fuel system is of sufficient quality and has sufficient performance capability to meet the reactor design requirements. Thus, data from this activity will 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 Requirements

Goals

The major goals of the PIE and safety testing activity are to:

- Collect relevant fuel PIE and safety testing data for the development and validation of fuel performance and fission product transport models and to demonstrate acceptable fuel behavior under normal operation and accident conditions as a function of temperature, burnup, fast fluence and coolant chemistry.
- Cooperate with other DOE programs and use international collaboration as much as possible to resolve key design and data needs and minimize duplication of effort.

Assumptions

Important activity assumptions are as follows:

- HTGRs will be designed such that the radionuclides are substantially retained in the core during normal operation and all design basis accidents.
- Radiologically significant reactivity transients are precluded by inherent characteristics of the design. Thus, no reactivity insertion accident testing is planned.
- Water ingress accidents are moderate rather than core flooding.
- Air ingress accidents are to be considered.
- DOE will implement the requisite cooperative agreements to facilitate cooperation with other DOE programs as well as international cooperation.

Requirements

The requirements for this activity are to:

- Confirm fuel performance under normal and accident conditions can be predicted to within the prescribed accuracy limits
- Collect the data to allow validation of the design methods used to predict fuel performance to prescribed accuracy limits in a manner acceptable to regulators
- Improve understanding of TRISO fuel behavior based on observed and measured phenomena that affect fuel performance and fission product release.

3.3.2 Scope of PIE and Safety Testing

In most cases, the major PIE and safety testing data needs are sufficiently well known and lead directly to the measurements or tests to be performed to satisfy those data needs. Whereas many of the needed facilities and apparatus exist from prior fuel development programs, restoration, modifications, and upgrades are necessary. In some cases, development of a new measurement technique may be required to satisfy a data need, which leads to a task to develop or apply that new technique.

HTGR fuel has been tested and examined at ORNL since the 1960s. The ORNL hot cells have a full range of capability to support the examinations that have been used in the past. INL hot cells have also been used to examine a wide variety of irradiated fuels for many years, including TRISO-coated Li target particles for tritium production in the NPR program. Both hot cell facilities were operating and functional at the beginning of this program. However, equipment refurbishment, modification or upgrading is necessary in some cases and procedure development and personnel training are necessary to meet NQA-1 requirements. Both laboratories have development staff capable of designing, procuring, and installing the equipment, and developing the protocols for new or additional examination methods required for the AGR program.

The tasks associated with PIE and safety testing are briefly discussed below. As noted earlier, all PIE tasks may not be required, depending on results as the activity proceeds, but costs are based on full PIE and safety testing to provide the maximum envelope for program planning purposes. Determination of the required tasks for a particular test train capsule will occur during preparation of the PIE and safety testing plan. Whether a full range of examinations is required for fuel irradiated under the AGR program depends on many factors, including the defective fuel fraction measured during manufacturing and the in-pile R/B measurements. If the fuel manufacture effort is successful, the fuel may have few defective particles (a fraction of exposed uranium $<10^{-4}$) [10^{-4} HM contamination is not low] and low in-pile R/B ($<10^{-6}$), in which case, less PIE may be required, primarily addressing metallic fission product distributions and coating layer behavior. Conversely, if the as-manufactured fuel appears to be of high quality, but the in-pile R/B is unexpectedly high (as was the case with the NPR fuel); the PIE effort will be expanded to locate the source of the failures. Cost and schedule estimates for conducting PIE and safety testing are provided in Section 4.

The following subsections discuss required preparations to conduct PIE and safety testing activities, list the test trains and briefly summarize the PIE and safety testing objectives of each test train. The subsections then discuss in detail the PIE and safety testing tasks and identify the subset of tasks to be performed for each of the test trains.

3.3.2.1 General PIE and Safety Testing, Assessment, and Facility Preparation

General PIE needs of the program involve test train handling, test train disassembly, including removal and disassembly of irradiation capsules, fuel and capsule component examination, fission product transport, and fuel failure fraction determination. Safety testing needs involve measurement of fission product release during heating of the irradiated fuel as well as fuel examination, fission product transport, and fuel failure fraction determination following the tests. Many of the facilities and equipment required for these tasks were in place at the beginning of the AGR fuel program, although restoration, upgrading, improvements, and new capabilities are necessary. Most of these tasks have been conducted in the past, and the available experience is being factored into the reestablishment of capability.

More than one DOE complex facility is capable of conducting at least some of these tasks or could develop equipment to perform these tasks outside their current capability. Thus, an early task for each test train will be to determine the best way, within cost and schedule constraints, to conduct the PIE and safety testing within the DOE complex. Known equipment deficiencies also exist and will need to be addressed to fulfill the needs of this program. The following six tasks address these needs for each test train.

PIE Site Task

Review the capabilities of candidate facilities, existing and new, for performing the separate PIE tasks and developing new equipment to perform new tasks not possible at present. Determine how these

facilities might be integrated, and consider the implications of transport and time delays that might impact analysis, cost, and schedule.

PIE Preparation Task

Prepare the selected facilities for the AGR PIE and safety testing. Inventory the PIE and safety testing capability, and note equipment that needs to be developed. Finally, prepare the necessary Environmental, Safety, and Health documentation for this task.

The facilities and apparatus required to perform the PIE and safety testing must be made ready, and in some cases, upgraded to meet current performance expectations. Resources have been allocated in the program to develop detailed costs for the requisite upgrades.

Generally, the nominal time to complete PIE and safety testing of an irradiation test train is about 3 years, assuming that facilities and personnel are available. However, the AGR irradiation schedule will result in multiple test train capsules undergoing PIE and safety testing at the same time. Expansion of the PIE capabilities at one site or the sharing of PIE and safety testing work at two sites will therefore be necessary to handle the workload. This need is most pressing for complex, time-consuming tasks such as the safety tests, which involve high-temperature heating for extended periods followed by detailed examination of the fuel. Thus, construction of additional safety test facilities and possibly an additional irradiated microsphere gamma analyzer may be necessary to handle the workload within the schedule.

Helium/Air/Steam Ingress Safety Testing Development Task

A fuel heating facility exists at ORNL capable of 1,800°C in a helium atmosphere. A second furnace for heating in helium to 1,800°C has been commissioned at INL to handle the workload in the AGR program. While early testing will be done in pure helium to shakedown the equipment and establish reliable test protocols, heating tests beyond AGR-2 will include impurity gases in the helium typical of that expected in an HTGR during a depressurized conduction cooldown. In addition, a fuel heating facility will be developed to extend the chemical environment capabilities to heating to 1,600°C in oxidizing atmospheres typical of air and moisture ingress events.

Coating Physical Properties Equipment Development Task

Develop tools to investigate irradiated coating physical properties, especially the structure and isotropy of carbon. Material properties of interest could include strength, density, microstructure, layer bonding, permeability, and elastic modulus.

Fuel Compact Reirradiation Equipment Development Task

Develop a method to reirradiate a fuel compact in a reactor so that the release of short-lived radioisotopes, including 8-day I-131 and 5-day Xe-133, can be measured in safety tests.

3.3.2.2 PIE and Safety Testing Scope of Activities

Test train PIE and safety testing is composed of several tasks chosen from a variety of options. Some of these tasks may be conducted in parallel, while others must be conducted sequentially. For example, a test train capsule must be opened before any work can be done with the fuel, so it is a serial task. Fuel compact deconsolidation can be a parallel task, because only a portion of the fuel is used for the task; the remainder of the fuel can proceed to other tasks, unrelated to deconsolidation. The actual grouping and relationships of the tasks will be detailed in a specific experimental plan; but for planning purposes, it may be assumed that PIE and safety testing will take about 3 years to complete, given no restrictions on

resources. The following tasks outline the options that are likely to be available for PIE and safety testing. The actual tasks that are projected to be performed for a particular test train are shown in Table 2.

Table 2. Test train PIE tasks.

Task Number	Task	AGR-1	AGR-2	AGR-3/4	AGR-5/6	AGR-7	AGR-8
PIE TASK-1	Test train receipt and visual inspection	X	X	X	X	X	X
PIE TASK-2	Test train gamma scanning	X	X	X	X	X	X
PIE TASK-3	Test train and capsule disassembly	X	X	X	X	X	X
PIE TASK-4	Component metrology	X	X	X	X	X	X
PIE TASK-5	Compact shipments to ORNL	X	X	X	X	X	X
PIE TASK-6	Graphite holder gamma scanning	X	X	X	X	X	X
PIE TASK-7	Fuel compact gamma scanning	X	X	X	X	X	X
PIE TASK-8	Melt and flux wire analysis	X	X		X	X	
PIE TASK-9	Thermocouple analysis	X	X		X		
PIE TASK-10	Thermal properties measurement	X	X		X		
PIE TASK-11	Capsule deposited fission products	X	X	X	X	X	X
PIE TASK-12	Radionuclide transport in irradiated specimens			X			X
PIE TASK-13	Micro-scale analysis of fuel compacts	X	X		X	X	
PIE TASK-14	Compact deconsolidation	X	X		X	X	
PIE TASK-15	Compact leach-burn-leach	X	X		X	X	
PIE TASK-16	Particle inspection and sorting	X	X		X	X	
PIE TASK-17	Burnup measurement	X	X	X	X	X	X
PIE TASK-18	Irradiated microsphere gamma analyzer	X	X		X	X	
PIE TASK-19	Micro-scale analysis of fuel particles	X	X		X	X	
PIE TASK-20	Safety testing – compact re-irradiation	X	X	X	X	X	X
PIE TASK-21	Safety testing – heating tests	X	X	X	X	X	X
PIE TASK-22	Waste handling	X	X	X	X	X	X
PIE TASK-23	Reporting	X	X	X	X	X	X

The PIE and safety testing tasks will have to be integrated with other AGR activities so that the tasks can be conducted in an efficient and time-effective manner. The primary goal is to ensure that the needed measurements and tests can be accomplished with the required accuracy. If this is impossible, the program needs early notification so that alternative actions can be taken. In particular, some data may prove to be very expensive to collect, and different approaches to modeling or fuel qualification may have to be explored.

In general, the following PIE tasks will be conducted, as appropriate, for the set of AGR test trains:

- *PIE TASK-1: Test train receipt and visual inspection.* The transfer and nuclear accountability documentation will be completed, and the hot cell will be prepared for the delivery of the cask containing the test train. The cask will be transferred from the ATR to the MFC Hot Fuel Examination Facility (HFEF), where photo-visual examination of the test train will be conducted.
- *PIE TASK-2: Test train gamma scanning*. The intact test train will be analyzed using a high-resolution gamma scan in the axial direction to help verify the position of test train internal components.
- *PIE TASK-3: Test train and capsule disassembly*. The test train and the capsules to be examined, will be disassembled using in-cell machine tools and jigs to remove the fuel compacts and internal components of experimental value.
- *PIE TASK-4: Component metrology*. The fuel compacts and internal capsule components will be visually and dimensionally inspected.
- *PIE-TASK-5: Compact shipments to ORNL.* Selected compacts will be packaged and shipped from INL to ORNL for concurrent PIE work.
- *PIE TASK-6: Graphite holder gamma scanning.* Each empty graphite holder will be scanned for fission products. 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.* Flux and melt wires will be removed from the graphite holders and analyzed to determine neutron flux levels and possible indications of high temperatures.
- *PIE TASK-9: Thermocouple analysis.* Selected thermocouples will be examined to identify any chemical interactions that could have implications for fuel coating interactions and to aid in future test train design.
- *PIE TASK-10: Properties of irradiated materials specimens*. Measure properties (thermal, physical, mechanical) as requested on samples of irradiated materials such as PyC, SiC, and graphite.
- *PIE TASK-11: Capsule deposited fission products.* The interior metal surfaces of each capsule will be analyzed for the presence of fission products that may be released from the fuel during irradiation.
- *PIE TASK-12: Radionuclide transport in irradiated specimens.* Measure radionuclide content and gradients in irradiated specimens using appropriately established techniques such as beta and gamma spectrometry and neutron activation.
- *PIE TASK 13: Micro-scale analyses of fuel compacts.* Selected compacts after irradiation and after accident testing will be analyzed in cross section at the microscopic scale to assess localized effects of irradiation and testing on the compact matrix and embedded fuel particles.
- *PIE TASK-14: Compact deconsolidation.* Selected compacts will be deconsolidated to free individual fuel particles from the matrix binder as a precursor to the leach-burn-leach process and to provide loose fuel particles for other PIE tasks.
- *PIE TASK-15: Compact leach-burn-leach.* An initial acid leach will be performed on deconsolidated particles to dissolve uranium and fission products in the matrix and exposed kernels. The particles

will then be exposed to air at elevated temperatures (e.g., 750°C) to remove pyrocarbon material not protected by intact SiC coatings. A post-burn leach will then be performed to dissolve uranium and fission products exposed by the burn step.

- *PIE TASK-16: Particle inspection and sorting.* Particles and deconsolidation debris from deconsolidation and/or leach-burn-leach will be examined with sufficient magnification to provide an indication of the condition of the particles, including coating damage, if any.
- *PIE TASK-17: Burnup measurement.* The primary means of burnup measurement will be by activity ratios determined from the compact gamma scans in PIE TASK-7. Destructive chemical analysis methods will be used on particles from selected compacts as a "benchmark" to calibrate the burnup determinations from the gamma scanning data.
- *PIE TASK-18: Irradiated Microsphere Gamma Analysis.* Individual particles will be gamma counted to quantify the inventories of selected fission products. The data will primarily be used to gauge the relative fission product retention in each of the analyzed particles and may also be used to screen for failed particles based on radionuclide inventories prior to performing other analyses
- *PIE TASK-19: Micro-scale analysis of fuel particles.* Particles identified in the previous tasks will be prepared for individual examination, including high resolution optical microscopy and scanning electron microscopy (SEM).
- *PIE TASK-20: Safety testing – compact re-irradiation.* Selected compacts will be reirradiated prior to safety testing, primarily to generate short-lived fission products including I-131 and Xe-133, so that iodine release during testing can be measured.
- *PIE TASK-21: Safety testing – heating tests.* Selected compacts will undergo heating tests, both isothermal and time-dependent temperature ramping, over the range 1,400–1,800°C. Gaseous fission product release will be measured continuously during the test, and metallic fission product release will be measured by analysis of deposition on condensation surfaces periodically replaced and analyzed for deposited isotopes. While early testing will be done in pure helium to shake down the equipment and establish reliable test protocols, heating tests beyond AGR-2 will include small levels of impurity gases in the helium typical of that expected in an HTGR during a depressurized conduction cooldown. In addition, a fuel heating facility will be developed to extend the chemical environment capabilities to heating to 1,600°C in an oxidizing atmosphere typical of air and moisture ingress events. It will be used to test some of the AGR-5/6 and AGR-7/8 fuel compacts.
- *PIE TASK-22: Waste handling*. Collect, package, and dispose of waste and spent fuel generated during conduct of the AGR PIE.
- *PIE TASK-23: Reporting*. Disseminate the findings, results, and problems of the PIE task in both formal and informal reports. Support program requests for specific information, clarifications, and impact assessments.

3.3.3 Test Train Specific PIE and Safety Testing

A preliminary assessment of the applicability of the detailed PIE and safety testing tasks defined above to 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 the subsections following Table 2. Costs and schedules for PIE and safety testing of each test train are provided in Section 4.

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

A primary purpose of AGR-1, the first test train to undergo irradiation and PIE, is to gain experience with multi-capsule test train design, fabrication, and operation and to reduce the chances of test train or capsule failures in subsequent test trains. It also provides data on irradiated fuel performance to support specification of the fuel to be qualified in later irradiation test trains, early NGNP licensing interactions, and development of a quantitative understanding of the relationship between fuel fabrication processes, fuel product properties, and irradiation performance. The specific PIE and safety testing tasks anticipated to be performed on this test train are identified in Table 2 (above). The individual task scopes are summarized in Section 3.3.2.2.

3.3.3.2 AGR-2: PIE of Fuel Performance Test Train

The AGR-2 PIE provide irradiated fuel performance data beyond the on-line R/B measurements for the large coater UCO and UO_2 fuel types discussed in Section 3.2.3.2, broadens options and increases prospects for meeting fuel performance requirements, and supports development of a fundamental understanding of the relationship between fuel fabrication processes, fuel product properties, and irradiation performance. The specific PIE tasks and safety testing tasks anticipated to be performed on this test train are identified in Table 2.

3.3.3.3 AGR-3/4: PIE of Fission Product Transport Test Train

The AGR-3/4 PIE measures fission product diffusivities in fuel kernels and fission product sorptivities in graphite materials for use in the upgrading of fission product transport models and overall codes. This PIE will focus on measurements of fission product inventories and concentration profiles in the graphitic components with a focus on a full mass balance to support fission product transport model development. The specific PIE and safety testing tasks anticipated to be performed on this test train are identified in Table 2.

3.3.3.4 AGR-5/6: PIE of Fuel Qualification Test Train

The AGR-5/6 PIE documents fuel integrity and to measure safety test performance to demonstrate compliance with statistical performance requirements under normal operation and accident conditions. The primary interest is to verify proper fuel performance. This PIE makes heavy use of the fuel heating apparatus. The specific PIE and safety testing tasks anticipated to be performed on this test train are identified in Table 2.

3.3.3.5 AGR-7: PIE of Fuel Performance Limits Test Train

The AGR-7 PIE measures the capability of the selected fuel to withstand irradiation and accidents conditions beyond the conditions in test train AGR-5/6 in support of plant design and licensing. The specific PIE and safety testing tasks anticipated to be performed on this test train are identified in Table 2.

3.3.3.6 AGR-8: PIE of Fission Product Transport Test Train

The AGR-8 PIE measures fission product releases from compacts with known quantities of failed particles for use in validation of fission product transport codes. This PIE makes heavy use of the fuel heating apparatus. The specific PIE and safety testing tasks anticipated to be performed on this test train are identified in Table 2.

3.4 Fuel Performance Modeling

A key product of the program is the development of validated fuel performance models. As discussed here, fuel performance modeling addresses the structural, thermal, and chemical processes that can lead to coated-particle failures. It considers the effect 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 is also modeled. Many groups have attempted to model the performance of coated-particle fuels.[19] These efforts have not resulted in a comprehensive model capable of predicting fuel performance with sufficient accuracy to directly facilitate fuel design or replace the need for comprehensive test data in a licensing application. There are many reasons why the modeling effort has not yet succeeded. The most significant reasons 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 for this work are to:

- Develop fuel performance models of coated-particle fuel (either UO_2 or UCO) that are more first-principle based that can be used to:
  - Guide current and future particle designs
  - Assist in irradiation and safety experiment planning
  - Predict observed fuel failures
  - 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 models and methods
- Benchmark these models/codes against U.S. and international irradiation and safety experiments, where possible.

The current effort by the modeling working group is 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.[20] 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 the development of candidate coated-particle fuel designs, (2) predicting the performance of coated-particle fuel during irradiation testing and post-irradiation heating, and (3) calculating fuel performance for AGR core designs during normal operation and hypothetical accidents. Development of fuel performance models requires fundamental understanding of potential failure mechanisms and how these mechanisms depend on the irradiation conditions and the materials constituting the fuel. Accurate fuel performance modeling will also require good materials properties and constitutive relations. Table 3 summarizes the key fuel failure mechanisms associated with

TRISO-coated-particle fuel and how these mechanisms depend on reactor service conditions and particle design and performance parameters. The following failure mechanisms under irradiation were considered: pressure vessel failure, cracking of IPyC, IPyC partial debonding, kernel migration, and fission product attack. Under accident conditions, the following were considered: fission product attack, SiC thermal decomposition, increase in SiC permeability/SiC degradation, oxidation of the SiC layer, and rapid energy deposition. Table 4 summarizes the important material properties required for accurate modeling under irradiation and accident conditions. The state of knowledge of the specific properties, their importance to modeling, and potential measurement techniques are listed.

The scope of this section is limited to activities needed to support fuel performance modeling. However, as indicated in Table 4, fission product release from the kernel and transport of fission products through the coating layers directly affects 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 and fission product transport will be combined as appropriate in this plan. In addition, some aspects of this work are being addressed in DOE Nuclear Energy Research Initiative (NERI) and International Nuclear Energy Research Initiative projects.

Table 3. (continued).

Table 3. Summary of coated-particle failure mechanisms.

Failure Mechanism	Reactor Service Conditions	Particle Design and Performance Parameters	Comments
Parameters that strongly influence the failure mechanism			
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 vs. 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	Will be more important at high burnup in LEU fuels because of the greater yield of noble metals, such as Ag, from plutonium fissions. More important during accident conditions.

Failure Mechanism	Reactor Service Conditions	Particle Design and Performance Parameters	Comments
Diffusive release through intact layers	Temperature Burnup Temperature gradient Time at temperature	Chemical state/transport behavior of fission products Microstructure of PyC and SiC	Could be more important at high burnup in LEU fuels because of greater yields of palladium from plutonium fissions and because of higher temperatures in future designs more important under accident conditions
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 <1,600°C). Expected to be important at ~2,000°C. Degradation observed at 1,800°C in coated particles was attributed to this mechanism, but may have been fission product attack instead.
Increase in SiC permeability/SiC degradation	Burnup Temperature Fluence	Microstructure of SiC(*) Diffusion(*) Buffer densification and cracking(*) 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 plutonium 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 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; uncertainty is large here. However, available data indicate that reactivity events are relatively benign in comparison to other technologies.

Note: (*) indicates a potential parameter.

Table 4. (continued).

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 properties are thought to depend on anisotropy	X-ray, laser Raman, optical
PyC irradiation-induced dimensional change	Reasonably well known as functions 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	Special specimens (e.g., 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	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	BISO 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 equiaxed). 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 nontrivial 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 classical brittle ring technique. Also can use axial compression of a cylindrical plug inside SiC cylindrical sample. Key issue is linkage of data to microstructure.

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 SiCf/SiC composites.	Critical to understanding the nature of debonding of the layers. The nature of the bond depends on the nature of the fabrication process.	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. More data in reactor-relevant temperature range (1,000 to 1,300°C) would be useful.	Lower importance given uncertainty in other parameters	Density (density gradient column) measurements
Irradiation-induced SiC creep	Limited data at low fluence.	Modest impact. PyC creep is much larger effect	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 the 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	Conventional techniques Small sample size adds to overall difficulty in measurement and ultimate uncertainty
Elastic modulus of PyC	Modulus is a function of anisotropy, fluence, density, and temperature. Few to no data at very high temperature expected in accidents.	Critical for potential reactivity events where large temperature gradients may develop within the fuel particle	Resonant ultrasound spectroscopy or nanoindentation

Property	Current State of Knowledge	Importance in Modeling	How to Measure
Elastic modulus of SiC	Data from fusion program show a 10% drop at reactor-relevant temperatures and radiation doses. Few data above 1,000°C.	Critical for potential reactivity events where large temperature gradients may develop with the fuel particle	Resonant ultrasound spectroscopy or nanoindentation
Thermal expansion coefficient of SiC	Limited number of data suggests expansion is constant between 900 and 1,300°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 with the fuel particle	Conventional techniques Small sample size adds to overall difficulty in measurement and ultimate 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 magnitude of effect
Buffer survivability	Failure of the buffer appears to be important to whether fission products get to the IPyC/SiC interface. This effect needs to be studied with the performance model before a definitive direction on the need for this work can be made.	We have some properties on buffer strength and dimensional change to determine its failure; these can be used as a starting point for evaluations	Would need to produce some low-density material for material tests
Kernel swelling under rapid energy deposition	Few data available under rapid energy deposition conditions for reactivity induced accidents that are more severe than anticipated for HTGRs. Under consideration under GIF VHTR fuel collaboration with Japan.	Kernel swelling and kernel-coating mechanical interaction may be critical to predicting failure in rapid reactivity transients	Part of PIE following reactivity transient testing

Note: (?) indicates a potential parameter.

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

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 is currently under way at ORNL and the University of Michigan. A large program is also under way in Europe as part of the GIF VHTR Fuel and Fuel Cycle Project Management Board collaboration activities.

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 is reasonably well understood. Fission gas release from UCO kernels is needed over the relevant burnup and temperature ranges for the NGNP. CO release from UO_2 is also needed at burnups in excess of 10% FIMA at relevant reactor temperatures (up to 1,300°C). Finally, measurements of kernel swelling for both UO_2 and UCO are needed, especially at high burnup. These measurements will be made for UCO as part of the AGR-3/4 irradiation and associated PIE.

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 (1,200–1,800°C). Also needed are the corresponding properties of SiC. The measurements will be made as part of the PIE.

3.4.4 Thermochemical Properties of Coating Layers under Accident Conditions

Fission products can interact with the SiC layer and degrade the properties of the layer. Of greatest concern is palladium attack under accident conditions. Many researchers have studied the attack of the SiC layer by palladium. The impact of the attack on the degradation of the properties of the layer has not been studied. Simple models assume that the particle fails when ~50% of the 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 Particle Fuel Model code (PARFUME). 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. Measurements of fission product attack will be made during PIE of fuel from the early AGR capsules.

Data from Germany suggest that the SiC layer becomes permeable to fission products under high-temperature heating when the coated particles are exposed to higher burnup and fast fluence conditions (e.g., 14% FIMA, 6 to 8×10^{25} neutrons/m^2). The permeability may be associated with a

microstructural change, or it 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. More evaluation of the original data is needed.

To evaluate the oxidation behavior of SiC, tests are planned as part of the accident heating tests in AGR-5 and 6 in which the influence of air on fuel behavior is studied. Low air partial pressures and fuel temperatures consistent with air ingress calculations will be used.

Kernel swelling and kernel coating mechanical interaction may be critical for predicting failure in reactivity-induced accidents and transients. These data can be obtained as part of PIE following reactivity testing. Reactivity testing is not part of the DOE AGR program because the likelihood of rapid (super prompt critical) reactivity transients that could induce fuel failures is precluded by design. Further input from plant designers and fuel vendors in conjunction with their interactions with the NRC is needed to resolve this question. Irradiated specimens will be archived for potential use by others in the event they proceed with this testing. Reactivity testing of unirradiated particles can be performed in Japan through the collaborative activities of the GIF/VHTR Fuel and Fuel Cycle Project Management board.

3.4.5 Thermophysical and Physiochemical Properties of Fuel Compacts

With the new fuel compacting process for AGR fuel, thermophysical and physiochemical properties of the compact need to be measured to enable accurate fuel performance assessments in the AGR irradiations and NGNP. 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 of AGR fuel.

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 accident conditions. The benchmarking of fuel performance codes is currently ongoing under the auspices of the IAEA for both normal and accident conditions through 2008, based mainly on historical irradiations and safety tests. Additional benchmarking is foreseen under the Generation IV VHTR Fuel and Fuel Cycle Program Management Board, based on the behavior of the current generation of TRISO fuel in current irradiations and safety tests planned in the U.S., France, European Union, South Africa, Korea, and Japan. Pretest predictions and post-test calculations will be performed for each irradiation in the program. Similar sets of calculations will be performed for a subset of the safety tests using accident performance models. In addition, as the new material properties data in the earlier tasks become available, the calculations will be rerun to understand the influence of the improved data on the predicted behavior. The performance test fuel and fuel qualification irradiations and accident testing, along with planned material property irradiations (obtained via NERI and international collaborations or by irradiation of material samples in HFIR at ORNL), will provide much of the separate effects data needed to improve the fuel performance models.

3.4.7 Code Verification and Validation

The verification and validation (V&V) of fuel performance codes are required for licensing. An independent integral test train irradiation (AGR-7) will be the key V&V activity for the normal fuel performance code. For safety work, a select number of heating tests of the irradiated fuel from this fuel performance validation irradiation will serve as a data set for V&V of the accident fuel performance model.

3.5 Fission Product Transport and Source Term

The goal of the Fission Product Transport and Source Term activity is to produce a technical basis for source terms under normal and accident conditions for the NGNP. The technical basis will be codified in design methods (computer models) that are validated by experimental data. The approach is to take credit for all fission product release barriers (kernels, coatings, graphite, primary coolant pressure boundary, and reactor building) to meet protective action guidelines at the exclusion area boundary with a vented low-pressure containment (VLPC) building. If one were to rely exclusively on the fuel particle coatings for radionuclide retention, the allowable failure fractions are reduced to about 2×10^{-7} for normal operation and about 2×10^{-6} for core heat-up accidents.[e] Such stringent limits on particle failure are considered impractical, at least for the foreseeable future, given that the best the highly successful German fuel development program could claim was 1×10^{-5} at 50% confidence and 5×10^{-5} at 95% confidence for normal operation of LEU UO_2 at 10% FIMA.

The testing and analysis activities outlined in this section are designed to produce validated fission product release models accurate to within a factor of 4 for fission gas and a factor of 10 for fission metal. The 4× and 10× values have been used previously[21] to guide the development of fission product behavior models that supported the gas cooled reactor designs in the 1980s and 1990s. The experimental work defined in this plan provides the technical basis for validating the design methods used to demonstrate that the reactor design meets the top-level radionuclide control requirements, including limits on occupational exposure and off-site doses, at the required confidence level. This will ensure that the 4× and 10× factors are appropriate for current PMR and PBR designs.

The goals, assumptions, and requirements for the fission product and source term work are presented below.

Goals

- Satisfy the fission product transport design data needs (DDNs) identified for the NGNP.
- Provide a technical basis for the source terms under normal and accident conditions.
- Reduce the uncertainty in the source term to less than the design margins of 4× for gases and 10× for metals.
- Validate design methods and codes for predicting source terms for normal and accident conditions.
- Utilize international collaboration to the fullest extent possible for the resolution of DDNs.

Assumptions

- HTGRs will be designed such that the radionuclides are essentially retained in the core during normal operation and all credible accidents.
- Full-scale prototype tests (e.g., a complete prismatic fuel element) and real-time tests are not required.
- HTGRs will be designed and licensed to operate with a VLPC building.

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e. For a 600-MW(t) GT-MHR operating on a LEU fuel cycle for which the I-131 core inventory is 16×10^6 Ci, assuming allowable I-131 release limits to meet the protection action guide thyroid dose limit of 5 rem are 2.6 Ci for short-term events such as a rapid depressurization, and 29 Ci for long-term events such as a depressurized core conduction cooldown.

- Some degree of credit must be taken for each of the principal fission product release barriers (kernels, coatings, matrix/graphite, primary circuit boundary, reactor building) to achieve a viable fuel and plant design and to provide the required defense-in-depth.
- TRISO fuel that meets fuel product specifications and in-core fuel performance requirements adopted for the HTGR can be mass produced at an acceptable cost.
- The accumulation of condensable radionuclides, especially silver and cesium isotopes, in the primary coolant circuits of direct-cycle HTGRs, have significant design, operations and maintenance, and safety implications.
- 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.
- Current interest in using high temperature steam for process heat applications implies the potential for moderate water ingress.
- The potential for air ingress into the primary coolant circuit of the HTGR exists, so its impact on fission product transport will have to be assessed.
- The current models and codes used to predict fission product transport have unacceptably large uncertainties. Model improvement and independent code validation will be required to ensure reliable plant design and licensing. (Fuel performance models, which are closely linked with fission transport models, are treated in Section 3.4.)
- Model development and code validation will be supported by the DOE AGR program.
- With few notable exceptions, the fission product transport data needs for direct-cycle HTGRs are largely generic, such that a base technology program can be defined that supports both prismatic and pebble bed core designs; moreover, there is considerable opportunity for international collaboration.
- Progress on resolving specific DDNs targeted for resolution in ongoing international programs in Russia, Europe, and Japan may be forthcoming and the essential data made available to the DOE AGR program in a timely fashion. DOE will implement the requisite cooperative agreements to facilitate this technology transfer.
- One or more loop tests may be needed to obtain definitive silver and cesium plateout data on relevant alloys under near prototypical HTGR conditions and to generate representative samples for decontamination studies.

Requirements

- Confirm the source terms under normal and accident conditions are accurate to within the prescribed accuracy limits.
- Validate the design methods used to predict fission product transport to the prescribed accuracy limits and to protocols acceptable to the NRC. (Fuel performance models closely linked with fission product transport models are treated in Section 3.4.)
- Generate an experimental data base to satisfy comprehensive set of fission product transport DDNs for the NGNP that defines the additional experimental data needed to confirm the source terms and validate the design methods.

3.5.1 Planned Tasks

The rationale for and a brief description of the planned tasks are provided in the program elements described in this section. Cost and schedule information is provided in Section 4.

R&D Task 3.5.1: Fission Gas Release, Including Iodine and Tellurium, from Failed Fuel Particles

The current database under normal operating conditions consists primarily of fission gas release measurements on laser-failed particles, including UCO kernels reirradiated in a training, research, and isotope reactor (built by GA) (TRIGA) reactor fitted with a high-temperature furnace. Isothermal in-pile hydrolysis tests on LEU UCO fuel have been performed at ORNL (HRB 17/18), and the temperature dependence of gas release from both unhydrolyzed and hydrolyzed LEU UCO fuel has been addressed in the Petten HFR-B1 test. The database contains results, under accident conditions, of heating laser-failed UC_2/ThO_2 particles, heating mechanically failed 20% FIMA LEU UCO particles from test HFR-B1, and integral data for 8–10% FIMA LEU UO_2 in the German heating program. Gaps in the information are fission gas release from failed LEU UCO particles under normal and accident conditions. This information is needed to improve the models in the SURVEY code for fission gas release under normal operating conditions and the SORS code for fission product release (gases and metals) under accident conditions. Sufficient single-effects data are needed to develop and refine gas release models with uncertainties <4× at 95% confidence.

R&D Subtask 3.5.1.1

Measure fission gas release (Kr, Xe, I, and Te) from exposed LEU/natural UCO kernels (compacts seeded at a known failure fraction in the range 0.001 to 0.01 with DTF particles, which are reference UCO kernels with a 10 to 15-µm PyC seal coat) irradiated in test trains AGR-3/4 under near normal HTGR flux over a range of temperatures. In addition to the results from AGR-3/4, prepare a report that brings together fission gas release data, currently scattered over many documents, generated in the HFR-B1 experiment carried out in Petten during the 1987–1989 time-frame. Update appropriate models in the SURVEY and PARFUME codes.

R&D Subtask 3.5.1.2

Measure fission gas, including iodine, from irradiated (test trains AGR-3/4), failed reference fuel under core conduction cooldown (CCCD) conditions. The atmosphere for the CCCD conditions will be determined as the core design and accident analyses evolve. Update appropriate models in the SORS and PARFUME codes.

R&D Task 3.5.2: Fission Metal Effective Diffusivities in Fuel Kernels

The fuel kernel of the coated particle is the initial barrier to the release of fission metals from the core and may provide significant holdup, especially in low-burnup kernels. Consequently, the transport properties of fission metals in LEU/natural UCO kernels must be characterized for normal operating conditions and for CCCD transients. The present database is derived primarily from measurements on particles irradiated in accelerated tests. There are German data for Cs, Sr, and Ag in UO_2 kernels of intact particles that were irradiated under near-real-time conditions, and limited laboratory data on cesium release from ThO_2 kernels as well. Gaps in the data that need to be filled are effective diffusivities of key fission metals (Cs, Ag, and Sr) and plutonium in LEU/natural UCO fuel kernels during normal operation and under CCCD transients. The data will be used to update the effective diffusivity correlations in the TRAFIC/COPAR code (used to calculate full-core metal fission product release under normal operating conditions), the TRAMP/COPAR code (used to calculate “hot spot” metal fission product transport under

normal operating conditions and for analysis of compacts in capsules in irradiation test trains), and the SORS code (used to calculate metal fission product release under CCCD transients). Sufficient single-effects test data are needed to develop and refine diffusivity correlations with uncertainties <10× at 95% confidence.

R&D Subtask 3.5.2.1

Measure and model fission metal release from LEU/natural UCO fuel kernels in failed particles under near real-time irradiation in test trains AGR-3/4. The DTF particles irradiated in seeded fuel compacts will be used. The principal information on metal release will be from mass balances derived from radiochemical measurements of the fission product inventories and concentration profiles with test articles of the capsules in the irradiation test train. Update the fission metal release correlations in the TRAFIC/COPAR, TRAMP/COPAR, and PARFUME codes.

R&D Subtask 3.5.2.2

Measure and model fission metal release from LEU/natural UCO fuel kernels in failed particles irradiated (in test trains AGR-3/4) in near real-time conditions and heated under CCCD conditions. The atmosphere for the CCCD conditions will be determined as the core design and accident analyses evolve. Data will be obtained by measuring time-dependent fission metal release at accident temperatures. Update the fission metal release correlation in the SORS and PARFUME codes.

R&D Task 3.5.3: Fission Product Effective Diffusivities in Particle Coatings

The fuel particle coatings, particularly the SiC coating, are the primary barriers to release fission products from the core during normal operation and during CCCD transients. The existing diffusivity correlations are largely inferred from particle release measurements for various fission products in SiC and PyC coatings in a laboratory environment. These data are supported by limited in-pile data for Cs, Sr, and Ag inferred from the results of irradiation experiments. Correlations are available for fission product diffusivities in SiC and PyC coatings derived from data taken on low-burnup German particles. It is recognized, however, that fission product diffusivities may depend on the structures of the SiC and PyC coatings, which vary with the coating manufacturing process. Therefore, fission product diffusivities need to be measured on particle coatings from U.S.-made TRISO fuel particles manufactured to specifications appropriate for AGRs. Sufficient data on the diffusivities of Ag, Cs, and Sr in SiC and I, Te, Xe, and Kr in PyC are required as a function of temperature, fast fluence, and as-manufactured coating attributes to reduce uncertainties to <10× at 95% confidence. Measurement of fission gas diffusivity in PyC has been reduced in priority (see Section 5.1.5).

R&D Subtask 3.5.3.1

Measure diffusivities of Ag, Cs, and Sr in SiC as a function of temperature, fluence, and as-manufactured coating attributes. Particles deconsolidated from irradiated (test trains AGR-1 and AGR-2) fuel compacts with low particle failure rates and screened by IMGA will be heated and diffusivities determined by measuring time signatures of fission product releases from the particles. Microprobe measurements of fission products found in coatings of intact particles from AGR-1 and AGR-2 will be used to determine if diffusivities for coating layers under irradiation can also be established.

R&D Subtask 3.5.3.2

Update correlations for fission metal diffusivities in SiC coatings in the TRAFIC/COPAR, TRAMP/COPAR, SORS, and PARFUME codes.

R&D Task 3.5.4: Fission Product Sorptivities/Diffusivities

Fuel-compact matrix and fuel-element graphite collectively constitute the third and final barrier to fission product release in a prismatic core (or simply the sphere matrix in pebble-bed cores). The matrix is relatively porous and provides little holdup of fission gases, including iodine, which is released from fuel particles and heavy-metal contamination. However, since it has a significant content of amorphous carbon, the matrix is highly sorptive of metallic fission products. The fuel-element graphite, which is fully graphitized, is somewhat less sorptive of fission metals than matrix, but it is more effective as a diffusion barrier than the latter.

Fuel element graphite can significantly attenuate the release of most fission metals, with the exception of cesium and silver, and preclude the release of actinides from the core during normal operation and CCCD transients. The present correlations for fission metal diffusivities in core graphite are derived largely from laboratory measurements on unirradiated nuclear graphites and from concentration profile measurements in various irradiated graphites. The correlations for Cs, Sr, and Pu sorptivities on graphite are derived largely from measurements on unirradiated graphites, but there are limited data for cesium and strontium on irradiated graphite and irradiated compact matrix material. The available data indicate that irradiation has relatively little effect on fission product sorptivities in the fuel matrix, but the transport of Cs, Sr, and Ag in graphite is strongly affected by neutron irradiation. The sorptivities of cesium and strontium on nuclear graphites have been shown to increase with increasing fast fluence, but the effect may anneal out at high temperature in the absence of a neutron flux. Limited laboratory data indicate that the vapor pressure of cesium over graphite increases in the presence of coolant impurities and as a consequence of partial graphite oxidation. Dragon Project data imply that silver transport through graphite may be reduced strongly at elevated system pressures. Sufficient single-effects test data are needed to develop and refine diffusivity and sorptivity correlations in fuel-compact matrix and fuel-element graphite with uncertainties <10× at 95% confidence.

Available sorption data for condensable radionuclides on primary circuit structural materials are deficient in several ways. There are few data for typical turbine blade materials such as Inconel 100 or hot duct/intermediate heat exchanger materials such as Inconel 617. Moreover, with the exception of tungsten data, these sorption measurements were made at partial pressures that are orders-of-magnitude higher than those predicted for the reactor during normal operation with high-quality fuel. The presence of surface oxidation is a confounding factor: cesium sorption on 304 stainless steel has been observed to increase by a factor of 40 to 60 when the surface was preoxidized, but iodine sorption on chromalloy steel decreased significantly when the surface was preoxidized. Sufficient experimental data are needed to characterize sorptivity of appropriate primary circuit structural materials at appropriate values of fission product partial pressures and representative oxidation conditions to within an uncertainty of <10× at 95% confidence.

Fission product deposition in the primary circuit could lead to high radiation dose rates that could interfere O&M operations. Therefore, the feasibility of decontaminating primary circuit structural materials needs to be investigated. The sorption measurements on primary circuit materials will produce representative contaminated specimens suitable for testing decontamination methods such as (a) conventional mechanical methods (e.g., high-pressure water and/or steam), (b) electrochemical leaching to chemically remove surface layers, and (c) commercial decontamination protocols developed for LWRs (e.g., low oxidation-state metal ion).

R&D Subtask 3.5.4.1

Measure sorptivities of Cs, Ag, Sr, and Pu in fuel-compact matrix and fuel-element graphites as a function of temperature, and as appropriate, coolant impurities, system pressure (for Ag), and the extent of graphite oxidation under normal operating and CCCD conditions. The sorption experiments will be conducted in a small-scale laboratory apparatus. Measure the diffusivities of Cs, Ag, Sr, and Pu in fuel-compact matrix and fuel-element graphites as a function of temperature and fast fluence in materials irradiated in test trains AGR-3/4.

R&D Subtask 3.5.4.2

Improve model for fission metal transport in graphite in the TRAFIC/COPAR, TRAMP/COPAR, SORS, and PARFUME codes.

R&D Task 3.5.5: Fission Gas Release Validation Data

The SURVEY and SORS codes require integral data on fission gas release for code validation that are independent of the data used to upgrade the code component models. The validation should ensure that the predictive methods are accurate to within 4× at 95% confidence. The validity of the SURVEY code has been assessed by using the code to analyze Fort St. Vrain (FSV), Peach Bottom, and several capsules in irradiation test trains. The noble gas release from FSV at the end-of-life was over predicted by about a factor of 2, where hydrolysis effects may have been smaller than in laboratory tests. The noble gas release from Peach Bottom Core 2 at the end-of-life was under predicted by a factor of 2 or 3; however, the dominant source of gas release was heavy-metal contamination. Both the FSV and Peach Bottom Core 2 contained (Th, U)C_2 fuel. Peach Bottom fuel was BISO-coated; FSV fuel was TRISO-coated, but the product specification allowed >10× higher as-manufactured coating defects than required for modern direct-cycle HTGRs. Fission gas release from capsules in irradiation test trains containing LEU UCO/ThO_2 fuel is generally predicted to within a factor of about 5. There is an inherent ambiguity in these data because the fuel failure fraction is not known with high accuracy, independent of the gas release data. Considerable gas release data from LEU UCO fuel were obtained in the COMEDIE BD-1 test. The validity of the transient gas release model in the SORS code used to analyze CCCD transients has not been rigorously assessed. The gaps are fission gas release measurements from LEU/natural UCO fuel with known failure under normal and accident conditions independent of data used to develop the SURVEY, SORS, and PARFUME codes.

R&D Subtask 3.5.5.1

Measure fission gas release (Kr, Xe, I, and Te) from LEU/natural UCO kernels with known fuel failure fraction (compacts seeded with missing buffer particles at a level of 10^{-3} to 10^{-2}) irradiated in test train AGR-8 under near normal HTGR flux over a range of temperatures for validation of the SURVEY and PARFUME codes.

R&D Subtask 3.5.5.2

Validate the SURVEY code for calculating fission gas release by performing pretest predictions and post-test calculations with the code and comparing the results with measurements.

R&D Subtask 3.5.5.3

Measure fission gas, including iodine, from irradiated (test train AGR-8) reference fuel with known failure fraction under CCCD conditions for validation of the SORS and PARFUME codes. The atmosphere for the CCCD conditions will be determined as the core design and accident analyses evolve.

R&D Subtask 3.5.5.4

Validate the SORS code for fission gas release by performing pretest predictions and post-test calculations with the code and comparing the results with the measurements.

R&D Task 3.5.6: Fission Metal Release Validation Data

The TRAFIC/COPAR, TRAMP/COPAR, and PARFUME codes for calculating fission metal release under normal operating conditions and the SORS and PARFUME codes for calculating fission metal release under CCCD transients require data for validation that are independent of the data used for code development. The validation should ensure that the predictive methods are accurate to within 10× at 95% confidence. The validity of the codes for predicting fission metal release under normal operating conditions has been assessed by applying them to predict the observed metal release in operating HTGRs (Peach Bottom Core 2 and FSV), and in irradiated capsules and in-pile loops. Most of the available data are for cesium, with a smaller number of silver and strontium data. In general, the releases of fission metals were under-predicted by factors of several, and in some cases, by more than an order of magnitude. A significant gap in the needed data is that only the COMEDIE BD-1 assessment was conducted on LEU/natural UCO fuel. The validity of codes for predicting fission metal release during CCCD transients has not been systematically assessed.

R&D Subtask 3.5.6.1

Validate the TRAFIC/COPAR, TRAMP/COPAR, and PARFUME codes by measuring fission metal release from irradiation test train AGR-8 containing LEU/natural UCO fuel with known failure under normal operating conditions.

R&D Subtask 3.5.6.2

Validate the TRAFIC/COPAR, TRAMP/COPAR, and PARFUME codes for fission metal release by performing pretest predictions and post-test calculations with the code and comparing the results with the measurements.

R&D Subtask 3.5.6.3

Validate the SORS and PARFUME codes by measuring fission metal release from irradiated (test train AGR-8) reference fuel heated under CCCD conditions. The atmosphere for the CCCD conditions will be determined as the core design and accident analyses evolve.

R&D Subtask 3.5.6.4

Validate the SORS and PARFUME codes for fission metal release by performing pretest predictions and post-test calculations with the code and comparing the results with the measurements.

R&D Task 3.5.7: Radionuclide Deposition Characteristics on Structural Metals

Condensable radionuclides, including iodine and volatile fission metals, released from the core during normal operation and during certain accidents will tend to deposit on structural metal surfaces within the primary coolant circuit, thereby attenuating their release to the environment. However, this plateout activity and the attendant radiation fields significantly complicate plant operations and maintenance, especially for a direct-cycle plant. Correlations currently available that describe the deposition behavior of condensable radionuclides on structural metals have very large uncertainties (»10×). A major cause of these large uncertainties is that the sorption isotherms were typically measured in the laboratory at partial pressures orders of magnitude higher than those that occur in the reactor; moreover, for cesium and silver,

the isotherms were measured on atypical materials (e.g., tungsten). The current database is inadequate to estimate the potential importance of diffusion of deposited radionuclides into the interior of structural metals ("in-diffusion") at operating temperatures. Data are needed to characterize the deposition of Cs, Ag, I, and Te on structural metals. Correlations are needed that give the sorptivities of these nuclides as a function of temperature, partial pressure, surface state, and coolant chemistry for normal operating conditions and under CCCD transients. Sufficient test data are needed to characterize the deposition, sorptivity, and diffusivity of cesium and iodine on high-temperature structural metals to within an uncertainty <10× at 95% confidence.

R&D Subtask 3.5.7.1

Measure the deposition characteristics of Cs, Ag, I, and Te on structural metals as a function of temperature, partial pressure, surface state, and coolant chemistry under normal operating conditions and CCCD transients in a series of out-of-pile loop tests. The atmosphere for the CCCD conditions will be determined as the core design and accident analyses evolve.

R&D Subtask 3.5.7.2

Measure sorptivities of Cs, Ag, Te, and I on primary circuit structural alloys as a function of temperature, partial pressure, surface state, and coolant chemistry in a small-scale laboratory apparatus. Determine to the extent practical the degree of penetration of the deposited radionuclides into the bulk metal and derive an effective diffusivity. Perform decontamination experiments on test specimens screening the decontamination techniques listed above.

R&D Subtask 3.5.7.3

Update correlations for fission product deposition on structural metals in the PADLOC code.

R&D Task 3.5.8: Radionuclide Reentrainment Characteristics for Dry Depressurization

Radionuclides that deposit in the primary coolant circuit during normal operation may be partially reentrained and released from the circuit during primary coolant circuit leaks. The correlations for predicting radionuclide reentrainment during dry depressurization transients contain very large uncertainties (»10×). The liftoff database was obtained in blow-down tests wherein the test specimens were mechanically removed from the loop or reactor in which the plateout activity was originally deposited. These ex situ blow-down data scatter badly and are not reproducible. The fractional liftoff of deposited activity was observed to be a function of the shear ratio (SR)—the ratio of the wall shear stress during the blow-down to that during normal operation—and, to a lesser extent, the duration of the blow-down. No correlation between the fractional liftoff and the blow-down temperature or the humidity of the helium was evident. Ex situ liftoff data from the Japan Atomic Energy Agency (JAEA) (small contaminated samples cut from a component of the Oarai Gas Loop-1 loop) suggest that reentrainment may be relatively modest, even for very large shear ratios such as SR >100. High-quality liftoff data were obtained in COMEDIE BD-1, for the materials of construction and service conditions of the steam-cycle MHTGR and the effect of dust was minimized by use of a full-flow filter (a planned second test with dust injection was not performed). The extent to which plateout activity may be removed during rapid depressurization transients must be quantified for HTGR materials of construction and service conditions. Correlations are needed for I, Sr, Cs, Te, and Ag as a function of SR, wall shear stress, blow-down duration, temperature, humidity, and surface oxidation state. Sufficient single-effects test data are needed to quantify the reentrainment characteristics of radionuclides deposited on structural metals to within an uncertainty of <10× at 95% confidence.

R&D Subtask 3.5.8.1

Measure reentrainment of the key radionuclides identified above under the conditions specified in a series of out-of-pile loop tests.

R&D Subtask 3.5.8.2

Improve the reentrainment model in the POLO code.

R&D Task 3.5.9: Plateout Distribution Validation Data

The PADLOC code, used to predict plateout distributions of condensable fission products in the primary coolant circuit, must be validated to have the specified accuracy (<10× at 95% confidence) for normal operating conditions and for CCCD transients. The data must be independent of those used to develop and upgrade the predictive methods. The accuracy of the current methods has been assessed by applying them to predict the plateout distributions observed in operating HTGRs (Peach Bottom and Dragon) and in-pile loops. The plateout distributions of cesium in Peach Bottom and of Cs, I, and Ag in Dragon were predicted to within a factor of 2 to 3; however, most of these data are for plateout at surface temperatures in the range 250–500°C. Considerable data on the plateout of key radionuclides under conditions representative of steam-cycle MHTGR conditions were generated by the COMEDIE BD-1 test and used to assess the validity of the PADLOC code. Silver and cesium data have been reported from the COMEDIE SR-1 test in which the loop was operated at higher temperatures (up to 800°C), but the data scatter badly. The accuracy of the current methods used to predict plateout under CCCD conditions has not been assessed for direct-cycle materials of construction. Integral test data are needed for condensable fission product (Cs, Ag, I, and Te) plateout on structural metal surfaces under normal operating conditions and CCCD conditions. The tests need to include relevant materials of construction and need to be performed under relevant HTGR service conditions; it is highly desirable to include a small, simulated turbine because there are no existing plateout data on rotating machinery. The effects of dust on the plateout distribution need to be quantified, especially for pebble-bed designs.

R&D Subtask 3.5.9.1

Measure radionuclide plateout from integral tests under normal operating and CCCD conditions in a loop. The effects of dust should be quantified. Two tests are planned: one under “clean” conditions, the other with dust added. Independent of other considerations, an integral in-pile test loop is the preferred facility for this work. The extent of additional testing needed to validate the models used to establish plateout rates of condensable fission products, (e.g., the need to conduct additional in-pile and out-of-pile fission product transport tests or other less complex tests to validate plateout models and codes) is an open issue that is still under discussion in the NGNP Project. In addition to cost and schedule issues, the allocation of uncertainty in the effectiveness of each of the five barriers to fission product transport and the extent to which each barrier is counted on to mitigate fission product transport and release must be taken into account. Resolution of this open issue is expected to occur as the plant design progresses through design and as agreement is reached regarding the scope of licensing basis events for the NGNP.

R&D Subtask 3.5.9.2

Validate the PADLOC code by performing pretest predictions and post-test calculations and comparing results with measurements. The analysis of COMEDIE SR-1 and the BD-1 data with the PADLOC code has been done (see Section 5.5)

R&D Task 3.5.10: Radionuclide Plateout and Reentrainment (Liftoff) Validation Data

The POLO code used to predict the liftoff of plated-out fission products during primary coolant leaks must be validated to ensure a predictive accuracy within 10× at 95% confidence. The data must be independent of those used to develop and upgrade the predictive methods. The present database for validation of radionuclide liftoff is extremely limited and does not explicitly account for the effects of dust. In the single in situ blow-down test of the Cadarache Pegase Loop 2/4 in-pile loop, <0.5% liftoff of the deposited activity was observed; however, the maximum SR was only 1.08. Moreover, the Cadarache Pegase Loop 2/4 loop contained an inordinate amount of metal oxide aerosol, so the data are likely to be biased toward high. Considerable additional liftoff data were generated in the COMEDIE BD-1 test, in which four in situ liftoff tests were performed at SRs ranging from 0.72 to 5.7. The effects of dust were not included in these tests. Integral test data are needed for liftoff of key radionuclides from deposits on primary circuit metals during rapid depressurization transients, including the effects of dust.

The presence of circulating and/or deposited particulate matter in the primary coolant circuit may alter the plateout distribution in the circuit during normal operation and the extent to which condensable radionuclides are released from the circuit during depressurization transients. The available data on the effects of dust on radionuclide transport in the primary coolant circuit are largely from reactor surveillance measurements from Peach Bottom, Dragon, AVR, and HTTR. Samples of deposited particulate matter were obtained from an FSV circulator and have been partially characterized at ORNL. An FSV plateout probe removed at end-of-life was examined at INL, but no particulate matter was detected on the probe filters. There are British data on the transport of metal oxide aerosols in AGRs, but there are no data on the effects of such aerosols on radionuclide transport. Also German data from measurements are made in the AVR. The considerable dust in the AVR resulted primarily from mechanical attrition of the circulating fuel spheres. Limited semi-quantitative data are also available from the GA deposition loop program. In one test, a quantity of graphite powder was added to the out-of-pile loop, and the result was to alter the plateout distribution of ^{137}Cs and ^{90}Sr and increase significantly (>10×) the amount of liftoff observed in ex situ blow-down tests. The data needed are measurements under representative conditions that elucidate the effects of dust on the transport of condensable radionuclides in the primary coolant circuit during normal operation and the reentrainment of these radionuclides during rapid depressurization transients. Sufficient data are needed to ensure that dust effects do not preclude validating design methods to predict fission product transport within the primary coolant circuit to within an accuracy of 10× at 95% confidence.

R&D Subtask 3.5.10.1

Measure radionuclide reentrainment (liftoff) from integral tests during rapid depressurization transients in a loop. The effects of dust should be quantified. Two tests are planned: one under “clean” conditions and the other with dust added. Independent of other considerations, an integral in-pile test loop is the preferred facility for this work. The extent of additional testing needed to validate the models used to establish reentrainment rates of condensable fission products, (e.g., the need to conduct additional in-pile and out-of-pile fission product transport tests or other less complex tests to validate liftoff models and codes) is an open issue that is still under discussion in the NGNP Project. In addition to cost and schedule issues, the allocation of uncertainty in the effectiveness of each of the five barriers to fission product transport and the extent to which each barrier is counted on to mitigate fission product transport and release must be taken into account. Resolution of this open issue is expected to occur as the plant design progresses and as agreement is reached regarding the scope of licensing basis events for the NGNP.

R&D Subtask 3.5.10.2

Validate the POLO code by performing pretest predictions and post-test calculations with the codes and comparing the results with the measurements.

R&D Task 3.5.11: Fission Product Transport in a VLPC

The VLPC is a significant barrier to the release of radionuclides to the environment during CCCD transients. The compartments and spaces in the reactor silo building are connected together to form a long and tortuous vent path. During events involving primary coolant leakage into the reactor building, natural processes will act to reduce the level of entrained radionuclides as the gas stream transits the building. Natural removal mechanisms, including condensation, gravitational settling, and turbulent deposition will attenuate radionuclide release by at least an order of magnitude. It is not necessary to take credit for the reactor building as a radionuclide release barrier to meet 10 CFR 100 dose limits. However, mechanistic radionuclide retention in the VLPC is considered when showing compliance with the protection action guide dose limits at the exclusion area boundary with source terms for CCCD accidents. Data are needed to develop and validate the methods describing the transport behavior of condensable radionuclides in the reactor building under wet and dry CCCD conditions.

No direct measurements have been taken of radionuclide removal from contaminated helium by condensation, settling, and plateout under conditions expected in the VLPC during a CCCD transient. There is an extensive LWR database on the behavior of radionuclides in steam-liquid water mixtures, and several major experimental programs have been conducted on the behavior of radionuclides in LWR containment buildings (e.g., the DEMONA tests in Germany and PHEBUS tests in France). These LWR data have been reviewed (see Section 3.5.1) and found not to be applicable to the VLPC. However, system codes developed for analysis of fission product behavior in LWR containments, such as MELCOR and MACCS, may be useful, providing MHTGR-specific input data are available. Data are needed for the condensation, settling, and plateout of I, Cs, Sr, Te, and Ag on reactor building materials of construction. The effects of temperature, coolant chemistry, surface state, and aerosol sizes and concentration must be treated explicitly. The chemical forms of the key radionuclides must be determined with particular attention to the effects of coolant chemistry on composition.

R&D Subtask 3.5.11.1

New DDNs need to be defined for characterizing radionuclide transport in the VLPC for the accident scenarios postulated for the NGNP. A plan needs to be developed, defining an experimental/ analytical program to satisfy the DDNs, and executed. The radionuclides of interest are I, Cs, Sr, Te, and Ag.

R&D Task 3.5.12: Decontamination Efficiency of Pressure Relief Train Filter.

A filter is placed in the piping downstream from primary coolant relief valves to decontaminate gases released through the relief valves before entering the VLPC during overpressure transients (e.g., large water ingress). Methods have been established and validated for calculating the decontamination factor for LWR containment filters for air streams at low temperatures. These methods must be validated for helium/air and helium/steam at high temperature.

R&D Subtask 3.5.12.1

Conduct laboratory tests of the pressure relief train filter to measure decontamination factor for key radionuclides. Select several candidate filter mediums, and test over a range of bed depths for the expected range of blow downstream conditions. The effects of temperature and coolant chemistry must be treated explicitly. The chemical composition of the key radionuclides (I, Cs, Sr, Te, and Ag) must also be

determined. Revise the predictive model as necessary, and use it to optimize the relief valve train filter design.

R&D Task 3.5.13: Tritium Transport

A test plan for characterizing tritium transport has been published[22] and forms the basis for this task. Four DDNs have been developed and a series of single-effects tests have been identified to provide the experimental bases for improving component models and material property correlations to describe H-3 transport behavior in HTGRs. The program is comprised of four elements: (a) measurement of H-3 release from irradiated TRISO particles; (b) measurement of H-3 sorptivities of irradiated core graphites; (c) measurement of H-3 release from irradiated B_4C pellets; and (d) measurement of H-3 permeation rates through candidate intermediate heat exchanger and steam generator metals.

R&D Subtask 3.5.13.1

Measure the fractional releases of tritium from failed and intact, irradiated TRISO particles as a function of temperature, time, burnup, irradiation temperature, and water partial pressure. The required test articles are irradiated LEU UCO TRISO particles; however, some testing may be done with irradiated LEU UO_2 particles if they were to become available earlier than irradiated UCO particles.

R&D Subtask 3.5.13.2

Measure H-3 sorptivities of reference fuel-compact matrix and core graphites as a function of temperature, H-3 partial pressure, fast fluence, and coolant chemistry. These sorption data shall be obtained at representative reactor partial pressures.

R&D Subtask 3.5.13.3

Measure fractional releases of tritium from B_4C granules as a function of temperature, time, fast fluence, irradiation temperature, and water partial pressure. The need to apply a PyC and/or SiC coating to the B_4C granules to improve oxidation resistance and tritium retention will be determined.

R&D Subtask 3.5.13.4

Measure H-3 permeabilities of candidate heat exchanger metals as a function of tritium concentration, metal temperature, hydrogen partial pressure, and surface oxidation state; these permeation data shall be obtained with representative reactor coolant chemistry to the extent possible.

R&D Task 3.5.14 Moisture Ingress

The current interest in using high-temperature steam for process heat applications has led to the inclusion of a steam generator in the primary helium coolant system in recent evaluations of design options for the NGNP. This design option brings along with it the risk of steam generator tube leaks, resulting in moisture ingress into the primary coolant system of the NGNP. Thus, there is an increased need to address the effects of moisture ingress on fuel behavior and fission product release, and a study to address the planning of moisture ingress research in the NGNP/AGR program has been initiated. Specific tasks will be defined as the planning study progresses.

A review of the considerable experimental and analytical work on moisture ingress accidents has recently been published.[23] This review treats analyses of a range of moisture ingress accidents for the MHTGR with a steam generator in the primary system as documented in the MHTGR Preliminary Safety Information Document .[24] The review also treats experiments on moisture ingress, both in-pile and out-of-pile post-irradiation safety tests, and the extensive analysis of the experimental data. Although detailed

models of fuel hydrolysis have been developed, for the purposes of assessing the consequences of moisture ingress accidents on fission gas release from the fuel, it may be sufficient to know that (a) hydrolysis affects only fuel particles with exposed kernels, (b) the gas release is dominated by the release of stored gas, and (c) the release of stored gas is independent of the type of gas, Xe or Kr, independent of the isotope of the gas, and the fractional quantity released is independent of the fuel chemistry, UCO or UO_2.

The principal results of the experiments on fuel hydrolysis can be reduced to a logarithmic plot of fractional release of stored gas as a function of partial pressure of water vapor, where data from both in-pile and out-of-pile experiments fall on the same curve.[23] Recent reexamination[25] of the analyses of moisture ingress accidents documented in the MHGTR Preliminary Safety Information Document[23] indicates that the reactor is scrammed in a few seconds (8 to 22s) after initiation of moisture ingress, whereas, minutes to hours are required to release fission gas from the fuel in the experiments. While the coolant flow in experiments tends to be much slower than in an operating gas-cooled reactor, the experimental results indicate that the vast majority of fission gases released by fuel hydrolysis in a moisture ingress accident will be generated after reactor shutdown.

Thus, it is unnecessary to repeat the in-pile experiments, which have been carefully analyzed, but instead, attention should be directed at confirming the earlier results[22] with NGNP fuel in out-of-pile safety tests at various moisture levels and temperatures. In addition, it would be desirable to measure the release of iodine (reirradiation required) and metallic fission products in these out-of-pile safety tests, as no reliable data for the releases of these materials under moisture ingress conditions currently exist. The fractional release rate of fission gases continually drops in post-irradiation tests. Thus for reliable data interpretation, each test shall contain a single moisture injection. This approach means that numerous irradiated fuel samples (compacts or elements) containing DTF particles that provide a known source of exposed kernels will be required for these tests. There are already great demands on the PIE of compacts containing DTF particles in the AGR-3/4 experiment, so a capsule containing DTF particles will be accommodated in AGR-5/6 to produce irradiated compacts containing DTF particles for post-irradiation safety heatup testing in moist atmospheres. Temperatures in the range of 800–1,300°C (corresponding to pressurized cooldown conditions) and up to 1,600°C (corresponding to depressurized conditions) may be appropriate. Partial pressures of water vapor in the range of 10 to 50,000 Pa are anticipated to capture behavior across a spectrum of water leaks. Out-of-pile scoping tests of the oxidation of various materials including graphite, matrix, coating layers, and depleted UCO bare kernels in a thermogravimetric apparatus to measure weight loss at various impurity levels and temperatures would be useful in guiding the development of experiments using irradiated material.

Another aspect of moisture relates to impurities in the helium coolant gas. The impurity levels are much lower than the moisture levels in the ingress experiments discussed above. Testing with appropriate levels of impurities will be included in irradiation tests AGR-3/4, 5/6, and 7/8. Design data needs for MHTGRs with steam generators, going back to 1987[26] specify in-pile testing with impure helium containing 12.6 Pa of H_2O and CO_2 and 31.5 Pa of CO. More recently, an analysis of impurities in operating gas-cooled reactors, including historical reactors such as Dragon, Peach Bottom-1, AVR, FSV, and the currently operating high temperature test reactor (HTTR), suggest that in a modern gas-cooled reactor impurities on the order of somewhat less than 1 Pa for H_2O and CO_2 and in the range 1-10 Pa for CO and H_2 could be expected[27]. Considering the massive amount of graphite and the high operating temperatures in the core of an HTGR, the two key impurities that will influence the oxygen potential are CO and H_2O. Therefore, the AGR-3/4, AGR-5/6, and AGR-7/8 irradiations will contain some capsules with representative quantities of CO (about 5 Pa) and H_2O (about 1 Pa) in the helium coolant gas.

R&D Task 3.5.15 Air Ingress

Air ingress into the core of a HTGR can possibly 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. In all cases, the reactor power has been shutdown, so all fission product releases associated with oxidation will occur under post-irradiation conditions, as opposed to operating conditions. In both a prismatic and a pebble bed HTGR, a graphite or matrix thickness of about 5 mm must be permeated before air or reaction products can have access to a fuel compact (prismatic design) or fuelled region of a spherical fuel element (pebble bed design). Specific tasks will be generated as the planning study on air ingress progresses, subject to specific design issues and the approach taken by reactor designers to mitigate the accident.

Results of experiments using unirradiated compacts and fuel particles in Japan and irradiated fuel spheres and fuel particles in Germany have been reported.[20] The Japanese results of weight change as a function of time indicate that combustion of carbonaceous materials in an unirradiated fuel compact is complete after 20 hours in flowing air at 1,400°C, revealing the SiC layer of the particles. After this duration, the particle failure fraction was determined to be 6.9×10^{-4}. After 54 hours in flowing air at 900°C, a particle failure fraction of 1.2×10^{-3} was measured. German results on irradiated spherical fuel elements (burnup of about 9% FIMA) indicated a particle failure fraction of 2.4×10^{-4} after 410 hours in flowing air at 1,300°C, 7.3×10^{-4} after 70 hours at 1,400°C and 1.2×10^{-3} after 140 hours at 1,400°C. Analysis of German time-dependent data on air oxidation of irradiated spherical fuel elements and unbonded, intact fuel particles suggests that two mechanisms with different activation energies may be operating over the temperature range of 1,300 to 1,500°C.[20] A thermodynamic analysis indicates that oxidation of SiC can produce either protective SiO_2 or volatile SiO depending on the temperature and oxygen partial pressure.[28]

Three areas of research are suggested by the foregoing experience. First, it is paramount to study the air ingress behavior as a function of break size and location which is under way as part of the NGNP methods program. The second research area involves safety tests conducted on compacts irradiated in graphite sleeves (to simulate the approximately 5-mm-thick web) or irradiated spherical fuel elements at various partial pressures of oxygen over a temperature range to be determined. An attempt should be made to measure the releases of iodine (reirradiation required) and fission metals in addition to noble gases. A supply of irradiated compacts/fuel elements is required but could be obtained from archived AGR-1 and AGR-2 compacts or compacts from AGR-5/6. Third, the transition from the formation of protective SiO_2 to the formation of volatile SiO should be mapped as a function of temperature and oxygen partial pressure to confirm the thermodynamic analysis. The use of surrogate kernels with the SiC layer exposed may simplify the experiment in this third area. As in the case of moisture ingress, scoping experiments in a thermogravimetric apparatus using unirradiated materials (compacts in a graphite sleeve or spherical fuel elements, in both cases containing fuel particles with surrogate kernels) and various partial pressures of oxygen and temperatures would be useful in guiding the development of safety tests conducted in oxidizing conditions on irradiated materials.

3.6 Other Activities

There are a few other activities in the AGR program that are accounted for in the cost estimate (in Section 4) separately because they do not fit easily into any one of the AGR experiments or they cut across the different work breakdown structure (WBS) elements in the program. These include:

1. Reports that document the results of the AGR program at key points in time to be given to the NGNP design and licensing organizations for development of topical reports or production of the safety documentation for the plant.
2. Facility upgrades required to accomplish irradiation and PIE activities. The experience in the AGR program to date has been that much of the infrastructure needed to carry out this fuel development and qualification program is in need of repair/upgrade or does not exist. These upgrades will enable the program to obtain the data outlined in the plan.
3. Upgrades to the NGNP data management and analysis software system (NDMAS) used to qualify and store all of the data generation in the AGR program to incorporate the latest versions of underlying software and interfaces with the internet that are anticipated over the next decade.

4. PROGRAM SCHEDULE AND COST

A detailed resource-loaded activity-based schedule for the activities presented in the 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 is through the irradiations early in the program and then shifts to the PIE and safety testing later in the program. Irradiation durations are determined by their location in the ATR. AGR-1 and AGR-2 are longer irradiations because of the lower thermal flux in the respective irradiation positions. AGR-3/4 through 7/8 are shorter irradiations because they will be located in a higher flux position in the ATR. The durations for PIE and safety testing are based on (a) estimates of throughputs at ORNL and INL based on the scope of anticipated activities considering historical and current experience in Germany and Europe for safety testing, (b) anticipated learning curve effects for the safety testing and PIE of later capsules and (c) schedule overlaps in the safety testing and PIE related activities for fuel from each of these capsules. Based on the schedule, the fuel for NGNP is anticipated to be qualified by mid-2022 assuming the funding levels required to accomplish the tasks is made available.

A detailed cost breakdown is shown by year in Figure 4. Fabrication, irradiation and PIE activities are grouped by irradiation (i.e., AGR-1, AGR-2). Separate cost lines are shown for fuel performance modeling and fission product transport scopes. Additional lines are provided for the other activities described in Section 3.6 since they cut across the program WBS elements. Costs in Figure 4 are actual costs up through FY 2010. Estimates are provided for activities in FY 2011 through FY 2022. In the table, the costs are also broken down by each of the major activities in the WBS. The total program cost is estimated to be ~ $340 M.

Figure 3. Fuel development and qualification schedule.

Activity	Start	Finish
FUEL DEVELOPMENT AND QUALIFICATION	10/1/03	9/4/20
AGR-1 - Shakedown Small Coater Fuel (Large B)	10/1/03	4/23/13
AGR-1 Fuel Fabrication	10/1/03	9/29/06
AGR-1 Design and Assembly	10/1/03	12/22/06
AGR-1 Irradiation	12/25/06	11/20/09
AGR-1 PIE and Safety Testing	11/23/09	11/6/12
AGR-1 Data Analysis and Reporting	10/1/03	4/23/13
AGR-2 Production Scale Fuel Performance (Large B)	10/2/06	10/4/16
AGR-2 Fuel Fabrication	10/2/06	4/9/10
AGR-2 Design and Assembly	5/5/08	6/11/10
AGR-2 Irradiation (Large B)	6/23/10	3/26/13
AGR-2 PIE and Safety Testing	3/27/13	4/19/16
AGR-2 Data Analysis and Reporting	10/2/06	10/4/16
AGR-3/4 Fission Product Transport Data (NE Flux Trap)	5/8/06	11/7/16
AGR-3/4 Fuel Fabrication	5/8/06	3/14/11
AGR-3/4 Design and Assembly	3/2/09	7/11/11
AGR-3/4 Irradiation	10/4/11	4/29/13
AGR-3/4 PIE and Safety Testing	4/30/13	5/25/16
AGR-3/4 Data Analysis and Reporting	5/8/06	11/17/16
AGR-5/6 Fuel Qualification (Flux Trap)	10/1/08	6/21/18
AGR-5/6 Fuel Fabrication	10/1/08	1/24/13
AGR-5/6 Design and Assembly	10/13/10	6/20/13
AGR-5/6 Irradiation	6/21/13	1/15/15
AGR-5/6 PIE and Safety Testing	1/16/15	12/22/17
AGR-5/6 Data Analysis and Reporting	10/1/08	6/21/18
AGR-7/8 Fuel Performance and Fission Product Transport V&V (Flux Trap)	7/2/12	9/4/20
AGR-7/8 Fuel Fabrication	8/1/12	1/30/13
AGR-7/8 Design and Assembly	7/2/12	6/14/13
AGR-7/8 Irradiation	8/10/15	3/3/17

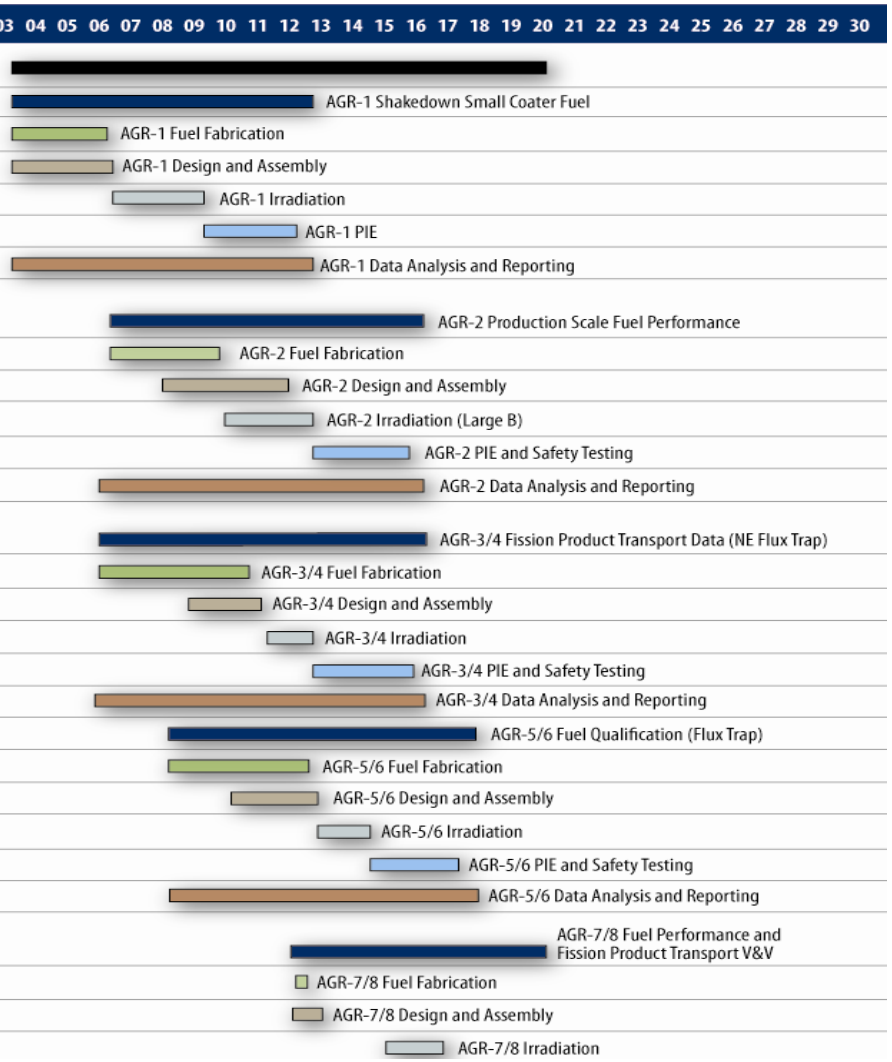

Figure 3. (continued).

Activity	Start	Finish
FUEL DEVELOPMENT AND QUALIFICATION	10/1/03	9/4/20
Fuel Performance Modeling	9/30/03	2/6/19
ZrC Modeling	10/1/07	9/30/13
PyC Modeling	10/1/07	8/22/14
AGR-1 Modeling	9/30/03	2/15/11
AGR-2 Modeling	3/26/10	6/18/13
AGR-3/4 Modeling	3/2/09	7/22/13
AGR-5/6 Modeling	4/19/13	4/9/15
AGR-7/8 Modeling	6/17/13	5/26/17
GIF VHTR Benchmarking	10/5/11	9/30/14
Special Fuel Studies	4/1/08	2/6/19
Fission Product Transport and Source Term	10/1/03	3/1/22
FPT and Source Term Planning FY03-FY08	10/1/03	10/1/08
Measure Sorption on Matrix and Graphite	1/3/11	3/5/20
Gas Loop Testing	10/3/11	12/20/21
Vented Low-Pressure Containment	10/1/12	2/1/16
H-3 Permeabilities of HX Metals	7/5/10	7/4/11
FPT and Source Term Data Qualification and Analysis	10/1/08	3/1/22
Supporting Activities	1/12/09	6/20/22
NRC Reports	9/9/13	6/20/22
Fuel Fabrication Commercialization	1/12/09	4/26/17
Key Milestone 1: Preliminary NRC Topical on Fuel Performance (AGR-1 PIE and AGR-2 Irradiation)	9/24/13	
Key Milestone 2: Final NRC Topical on Fuel Performance (AGR 3,4,5 and 6)	12/20/18	
Key Milestone 3: VHTR Source Term Validated	6/20/22	
Key Milestone 4: Fuel Performance Model Validated	8/17/18	

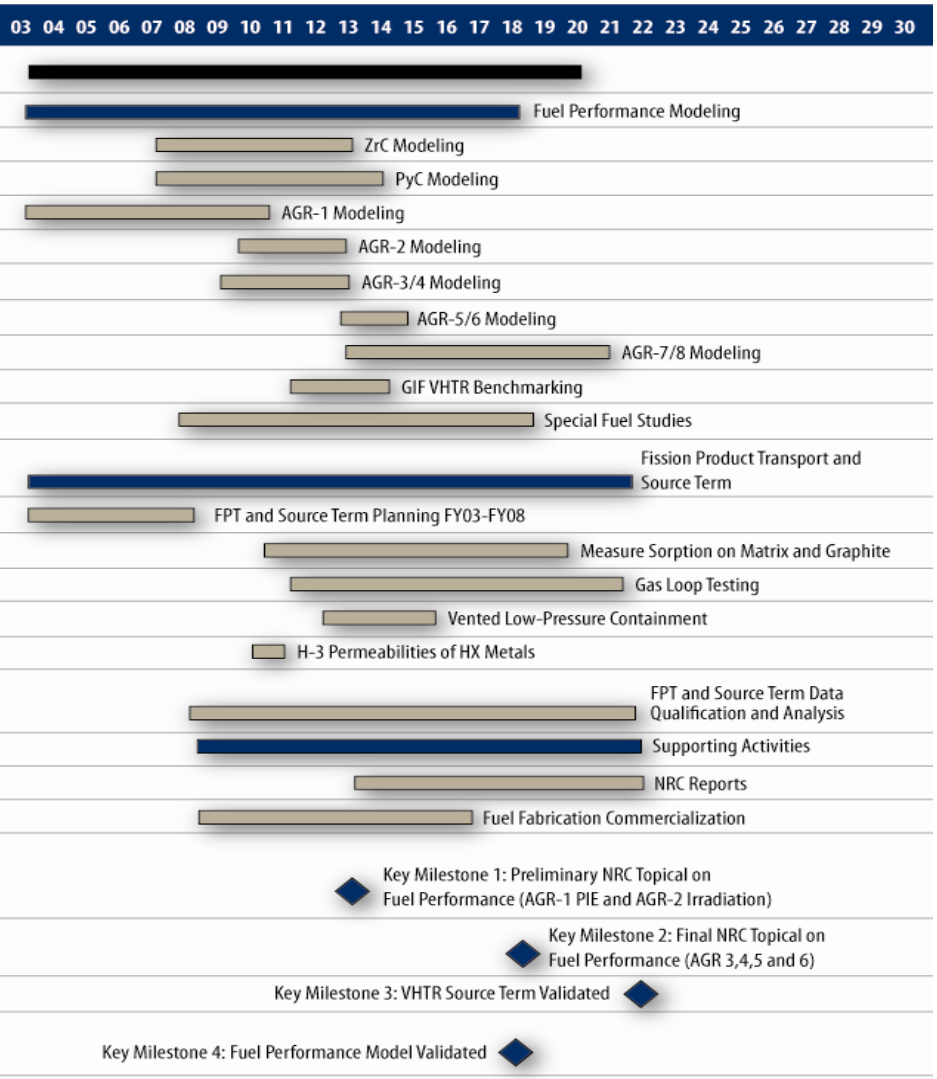

Figure 4. Fuel development and qualification cost schedule.

Fuel Development, Irradiation & PIE	FY03	FY04	FY05	FY06	FY07	FY08	FY09	FY10	FY11	FY12	FY13	FY14	FY15	FY16	FY17	FY18	FY19	FY20	FY21	FY22	Total
AGR-1 - Shakedown Small Coater Fuel (Large B)																					
Fuel Fabrication	$450	$3,185	$2,594	$4,602	$233	$35	$67	$100													$11,266
Design & Assembly	$66	$696	$62	$1,370	$108	-$29															$2,273
AGR-1 Irradiation			$2,252	$2,110	$1,311	$1,832	$1,745	$1,325													$10,576
AGR-1 PIE				$215	$248	$3,101	$7,252	$8,821	$5,706	$5,706	$951										$32,000
AGR-1 Data Analysis and Reporting							$2,256	$1,475	$439	$439	$256										$4,865
AGR-1 Total	**$516**	**$3,881**	**$4,909**	**$8,298**	**$1,901**	**$4,939**	**$11,320**	**$11,720**	**$6,145**	**$6,145**	**$1,207**	**$0**	**$0**	**$0**	**$0**	**$0**	**$0**	**$0**	**$0**	**$0**	**$60,981**
AGR-2 Production Scale Coater Development (Large B)																					
AGR-2 Fuel Fabrication					$2,113	$6,660	$2,102	$745													$11,620
AGR-2 Design & Assembly						$212	$1,231	$2,001													$3,445
AGR-2 Irradiation								$279	$893	$893	$447										$2,513
AGR-2 PIE & Safety Testing											$2,889	$5,502	$5,502	$3,210							$17,102
AGR-2 Data Analysis and Reporting								$265	$553	$553	$667	$475	$429	$429							$3,371
AGR-2 Total	**$0**	**$0**	**$0**	**$0**	**$2,113**	**$6,872**	**$3,333**	**$3,291**	**$1,446**	**$1,446**	**$4,003**	**$5,977**	**$5,931**	**$3,638**	**$0**	**$0**	**$0**	**$0**	**$0**	**$0**	**$38,051**
AGR-3/-4 Fission Product Transport Data (NE Flux Trap)																					
AGR-3/-4 Fuel Fabrication			$685	$120	$354	$206	$187	$1,079	$829												$3,460
AGR-3/-4 Design & Assembly						$5	$185	$831	$3,065												$4,086
AGR-3/-4 Irradiation								$120		$2,417	$1,430										$3,967
AGR-3/-4 PIE & Safety Testing											$1,595	$3,544	$3,544	$2,363							$11,046
AGR-3/-4 Data Analysis and Reporting									$269	$998	$987	$333	$333	$333							$3,253
AGR-3/4 Total	**$0**	**$0**	**$685**	**$120**	**$354**	**$212**	**$372**	**$2,030**	**$4,163**	**$3,415**	**$4,012**	**$3,877**	**$3,877**	**$2,696**	**$0**	**$0**	**$0**	**$0**	**$0**	**$0**	**$25,813**
AGR-5/-6 Fuel Qualification (Flux Trap)																					
AGR-5/-6 Fuel Fabrication							$6,608	$7,305	$4,932	$4,932	$1,890										$25,666
AGR-5/-6 Design & Assembly									$1,449	$1,449	$1,051										$3,948
AGR-5/-6 Irradiation											$604	$2,417	$806								$3,827
AGR-5/-6 PIE & Safety Testing													$2,719	$4,079	$4,079	$1,258					$12,136
AGR-5/-6 Data Analysis and Reporting									$157	$286	$404	$780	$594	$372	$372	$279					$3,245
AGR-5/6 Total	**$0**	**$0**	**$0**	**$0**	**$0**	**$0**	**$6,608**	**$7,305**	**$6,538**	**$6,667**	**$3,949**	**$3,197**	**$4,119**	**$4,452**	**$4,452**	**$1,537**	**$0**	**$0**	**$0**	**$0**	**$48,823**
AGR-7/-8 Fuel Performance and Fission Product Transport V&V (Flux Trap)																					
AGR-7/-8 Fuel Fabrication										$415	$790										$1,204
AGR-7/-8 Design & Assembly										$410	$1,969										$2,379
AGR-7/-8 Irradiation													$404	$2,426	$1,011						$3,841
AGR-7/-8 PIE & Safety Test															$2,246	$4,146	$4,146				$10,537
AGR-7/-8 Data Analysis Reporting													$140	$843	$662	$364	$364	$364			$2,736
AGR-7/8 Total	**$0**	**$0**	**$0**	**$0**	**$0**	**$0**	**$0**	**$0**	**$0**	**$825**	**$2,759**	**$0**	**$544**	**$3,269**	**$3,918**	**$4,509**	**$4,509**	**$364**	**$0**	**$0**	**$20,697**
Fuel Performance Modeling Total	**$148**	**$371**	**$711**	**$620**	**$178**	**$661**	**$1,192**	**$1,162**	**$2,377**	**$2,422**	**$2,494**	**$1,978**	**$757**	**$590**	**$540**	**$390**	**$143**	**$0**	**$0**	**$0**	**$16,734**
Fission Product Transport & Source Term Total	**$0**	**$82**	**$46**	**$71**	**$53**	**$396**	**$714**	**$654**	**$957**	**$4,323**	**$8,805**	**$9,231**	**$9,231**	**$7,468**	**$6,025**	**$6,025**	**$5,174**	**$3,971**	**$3,618**	**$1,327**	**$68,171**
NRC Reports Total	**$0**	**$0**	**$0**	**$0**	**$0**	**$0**	**$0**	**$0**	**$0**	**$0**	**$0**	**$419**	**$419**	**$419**	**$419**	**$978**	**$347**	**$0**	**$0**	**$496**	**$3,498**
Fuel Fabrication Commercialization Total	**$0**	**$0**	**$0**	**$0**	**$0**	**$0**	**$0**	**$0**	**$1,442**	**$3,528**	**$3,097**	**$758**	**$586**	**$839**	**$563**	**$0**	**$0**	**$0**	**$0**	**$0**	**$10,814**
Facility Upgrades Total	**$0**	**$0**	**$0**	**$0**	**$0**	**$2,309**	**$3,810**	**$991**	**$2,267**	**$2,267**	**$2,267**	**$2,267**	**$2,267**	**$2,267**	**$2,267**	**$2,267**	**$2,267**	**$2,267**	**$2,267**	**$0**	**$32,046**
NDMAS Upgrades								**$1,312**	**$522**	**$522**	**$522**	**$522**	**$522**	**$522**	**$522**	**$522**	**$522**	**$522**	**$522**		**$7,051**
PM Oversight and Technical Integration Total	**$592**	**$937**	**$1,077**	**$1,433**	**$645**	**$1,648**	**$1,331**	**$1,411**													**$9,074**
Grand Total	**$1,256**	**$5,270**	**$7,426**	**$10,542**	**$5,244**	**$17,038**	**$28,681**	**$29,876**	**$25,858**	**$31,560**	**$33,115**	**$28,226**	**$28,253**	**$26,159**	**$18,706**	**$16,228**	**$12,961**	**$7,123**	**$6,407**	**$1,823**	**$341,753**

FY03-FY09 TOTAL Actuals
FY10 Actuals thru 9/9/10
FY11-FY22 Projected Costs based on scheduled activities (includes PM Oversight and Technical Integration)

Figure 4. (continued).

Costs for Major R&D Areas	
Fuel Fabrication	$53,217
Design & Assembly	$16,132
Irradiation	$24,724
PIE & Safety Test	$82,822
Data Analysis	$17,470
Fuel Performance Modeling	$16,734
Fission Product Transport	$68,171
Other	$62,483
TOTAL	**$341,753**

5. ACCOMPLISHMENTS AND STATUS

The accomplishments and status of NGNP/AGR fuel development and qualification as of the time of this revision are discussed below by program element.

5.1 Fuel Fabrication

The progress made in the fuel fabrication area during the first 7 years of the program and the process development activities that remain to be completed to establish coated-particle fuel technology for the NGNP in the U.S. are discussed in the following subsections.

5.1.1 Prepare Irradiation Test Fuel Specifications

In the absence of a core design for the NGNP, the fissile fuel particle for the GT-MHR design[6] was selected as the fuel particle design for early fuel process development and for the initial irradiation test of AGR-1.[29] This fuel particle design consisted of a standard TRISO coating (as described above) deposited on a 350-µm UCO kernel having a nominal ^{235}U enrichment of 19.9%. This approach effectively addressed the multiple objectives of the AGR-1 irradiation test (see Section 3.2) and the need to begin the AGR-1 irradiation as soon as possible, given that AGR-1 was a critical path element of the program. Also, in the absence of a fuel form selection or an NGNP core design, the AGR fuel program has assumed that the fuel particles will be agglomerated into cylindrical fuel compacts similar to compacts in the GT-MHR design. A fuel particle packing fraction of ~35% and a compact length of 1 inch (as opposed to the GT-MHR design of about 2 inches) were selected for the AGR-1 and AGR-2 irradiation test compacts in order to ease the burden on initial compact process development and to limit the schedule risk associated with the irradiation tests. The feasibility of higher packing fractions will be investigated further, but the length of the compact will remain at 1 inch unless future data indicate a significant economic advantage for a longer compact (high volume automation economics may favor the 1 inch compact). A preliminary fuel product specification[30] for the AGR fuel program was prepared based on the GT-MHR Fuel Product Specification[6] to provide guidance to the fuel process development effort.

A fuel form had not been selected for NGNP at the time of specification of fuel for the AGR-2 irradiation, and a detailed core design was unavailable, thus both UO_2 particles based on the German pebble bed design and UCO particles based on available reactor physics and thermal analysis results for a prismatic core[31,32] are included in AGR-2, with both particle types irradiated in compact form. The specification of the UCO particle design for the AGR-2 irradiation is based on available prismatic NGNP core design study results that use a single fuel particle design. This creates the need to develop and qualify a fissile fuel particle having a larger diameter kernel and lower ^{235}U enrichment than used in AGR-1. On this basis, a kernel diameter of 425 µm and an enrichment of 14% have been selected for UCO fuel particles in AGR-2. Detailed fuel product specifications were prepared to define the specific fuel requirements for the AGR-1,[16] AGR-2,[33] and AGR-3/4[34] irradiation tests.

5.1.2 Fuel Kernel Manufacturing

B&W used its reference internal gelation process to fabricate ~6 kg of NUCO kernels for use in initial coating process development in a small coater,[35] ~4 kg of LEU UCO kernels for use in AGR-1 fuel fabrication,[36] and 0.5 kg for use in fabrication of driver fuel for the AGR-3/4 fission product transport irradiation test.[37] Fabrication of the NUCO kernels and LEU UCO kernels was completed in 2004 and 2005, respectively. However, unexpected problems were encountered in producing both the NUCO and LEU UCO kernels. These problems included achieving the specified density and kernel chemistry, large within-batch and batch-to-batch variability in product properties, and kernels with poor mechanical

strength. Substantial effort and funds beyond what was anticipated in the initial program planning were dedicated to resolving these problems, but the initial results were not completely satisfactory in meeting long-term program needs. The kernel fabrication process was improved and stabilized to the point where it was possible to fabricate an acceptable LEU UCO kernel composite for use in fabricating fuel for AGR-1, but the need for further process development to improve process repeatability and the uniformity and quality of the product was evident.

Consequently, a further UCO kernel process improvement effort was executed. The scope of this task included the following elements:

- Carbon dispersion in the ADUN
  - Optimize carbon source and surfactant
  - Investigate carbon dispersion in the hexamethylene tetramine (HMTA), as opposed to the ADUN
  - Optimize ultrasonic parameters and shelf-life of carbon dispersion
  - Evaluate alternate schemes for carbon dispersion
  - Develop specifications for the carbon source and dispersant used in UCO kernel fabrication
  - Develop QC procedures to support kernel fabrication at each step of the process
- Sintering process
  - Optimize carbothermic reduction step and kernel densification conditions
  - Establish conditions that eliminate sticking of kernels in the sintering furnace
  - Determine reactions that occur during the calcining step
  - Develop a low-temperature sintering scheme
- Reduce product variability
  - Explore influence of changes in ADUN and HMTA flow rate during a kernel forming run-on kernel variability
  - Analyze broth pot mixing
  - Determine ideal broth zone for UCO kernel formation.

The development effort described above was completed and highly successful in improving UCO kernel properties, reducing property variability, simplifying process steps, and increasing throughput rate.[38,39,40,41] Three composites of 425 µm NUCO kernels totaling 37 kg were then fabricated for use in large-coater process development.[16] A smaller composite of 425 µm LEU UCO kernels (~7 kg) and 500 µm LEU UO_2 kernels (~10 kg) were fabricated for use in making fuel for AGR-2. Additional qualification test fuel for AGR-5/6, fuel for the fuel performance validation irradiation (AGR-7), and fuel for the fission product transport validation irradiation (AGR-8) will be produced. Because of the uncertainties noted regarding the NGNP core and fuel designs, changes to this kernel procurement strategy may be required once a NGNP core design is available. Resource and schedule uncertainties with regard to kernel fabrication will thus remain until detailed core and particle design results become available.

Incremental improvements in throughput rates have been achieved during the production of approximately 80 kg of NUCO kernels during FY 2009 and FY 2010, along with improvements in product yields.[42] The current capacity of the B&W pilot kernel-fabrication line is about 120 kg U per year based on a single shift per day, 5-day per week, and 40 week per year operation. The rate is limited by the forming line, specifically washing and drying the kernels. With relatively minor modifications to the forming line, and assuming the same operating basis, a rate of 240 kg U per year is achievable. The current B&W facility is primarily designed as a development facility and is licensed to handle all uranium enrichments (from depleted to fully enriched materials); the ability to process highly enriched uranium material limits the throughput capacity of the facility. However, the key components in the system are essentially full-scale equipment, which could be used in a production facility. For example, the forming system currently used would be handled in a continuous rather than batch mode and would have a multiple-nozzle feed. The product collection system would need to be modified for the larger output, but only in the number of available receivers, and not necessarily in the receiver design. The current sintering furnace, if used in a three shift operation, could produce up to 5 runs per week and is capable of charge sizes of 5 kg or more if used in an LEU facility with appropriate nuclear criticality-safety controls. This would permit up to 1,000 kg U per year from one such furnace. The base systems being developed under this program are readily scalable to a full-size kernel production facility with capacity in ranges cited above.

5.1.3 Coating Process Development

The path to successful coated-particle manufacturing selected by the AGR program as having the lowest risk was to develop a coating process that replicates, to the extent possible, the properties of the coatings on the German fuel particles that have exhibited excellent irradiation performance. The following approach was taken to replicate the German coating properties:

1. Use coating process information from previous U.S. and international programs, including published information from the German HTGR Program,[f] to establish an initial reference set of process conditions and product specifications.
2. Perform coating experiments to better establish the relationships between coating properties and coating conditions for the specific coater in use. Make adjustments to the initial reference set of process conditions, as necessary, based on the results of these coating experiments.
3. Fabricate coated particles using the reference set of process conditions.
4. Compare the properties of the coatings with those on archived German particles to determine whether coatings with equivalent coating properties have been achieved. Make adjustments to the reference set of process conditions based on these property comparisons.
5. Iterate Steps 3 and 4 until coatings are achieved that have properties equivalent to the properties of the German coatings.

Economical coated-particle production is a function of the coater size. Achievement of the overall program goals will, therefore require qualification of fuel fabricated in a “production-size” coater. But coating process optimization using a production-size coater would be very expensive and require handling substantial quantities of enriched material. The initial coating process development effort, therefore, involved experimental work in a 2 inch-diameter, laboratory-scale coater at ORNL to (1) resolve

f. The program decided not to use German coater design and coating process data privately held by GA to avoid intellectual property issues or royalties related to these data.

questions concerning the feasibility of the uninterrupted German coating process for coating the smaller LEU UCO kernels (350-µm diameter UCO versus the 500-µm-diameter UO_2 used in the German Program), (2) develop a better fundamental understanding of the coating process and the relationship between coating properties and coating process conditions, and (3) fabricate fuel for the AGR-1 irradiation test. The key issues, as defined at the beginning of the AGR program, concerning German coating process feasibility to be resolved by the small coater work included the following:

- *IPyC coating permeability vs. anisotropy*. The German process conditions deposits the IPyC coating result in a highly isotropic IPyC, but these conditions are suspected to also result in a somewhat permeable IPyC coating based on U.S. coating experience. The suspected permeability of the German IPyC coating is apparently acceptable when coating German UO_2 kernels, but it was considered a potential problem when coating UCO kernels because UCO kernels may be more vulnerable to be attacked by HCl gas (a byproduct of the thermal decomposition of methyltrichlorosilane) during SiC coating. If the IPyC layer is sufficiently permeable to allow HCl gas to react with the kernel to form volatile uranium chlorides during SiC coating, this could result in uranium dispersion into the buffer and IPyC coatings, and in a higher level of as-manufactured SiC defects. Therefore, it was necessary to determine whether the UCO fuel particles on which the IPyC coating layer is deposited using process conditions known to yield highly isotropic coatings were characterized by excessive uranium dispersion and/or SiC defects. Following an extensive investigation of IPyC coating conditions and resulting properties, a range of coating conditions were identified that produce acceptable permeability, uranium dispersion, and SiC defect fractions.[43]
- *Smaller size of LEU UCO kernels*. Practicable batch size and fluidization gas flows for the smaller UCO kernels in the laboratory coater were determined in the early stages of coating process development.
- *Uninterrupted coating*. In the German coating process, the buffer, IPyC, SiC, and OPyC coating layers are deposited in succession without unloading the fuel particles from the coater. With 500-µm UO_2 kernels, the volume of the particles increases by a factor of 6 during the coating sequence. For 350-µm kernels, the volume of the particles increases by a factor of 11. The feasibility of an uninterrupted coating sequence with the smaller kernels and the process conditions necessary to accommodate this larger increase in particle volume was demonstrated.
- *Lower SiC deposition temperature*. The SiC deposition temperature (~1,600°C) that has been used historically in the U.S. HTGR program produces large columnar grains oriented in the direction of deposition. Reduction of the deposition temperature by 50 to 100°C results in a microstructure characterized by smaller equiaxed grains believed to provide better retention of metallic fission products such as ^{137}Cs and ^{110m}Ag. Also, a lower SiC deposition temperature should reduce the potential for uranium dispersion during SiC coating, which may explain why Germany was able to deposit an IPyC layer that is relatively permeable compared to the IPyC layers deposited in the U.S. MHTGR program. The beneficial effect of the lower SiC deposition temperature on the IPyC coating permeability, the IPyC anisotropy, and the production of finer grained SiC microstructure was demonstrated.

The small-coater work also included characterization of archived German[44] and U.S.[45] fuel, and initial development of a computational fluid dynamics coating process model with the goal of predicting the impact of changes in process conditions on coated-particle properties and quality.[46]

The work outlined in the logic diagram in Figure 5 to establish and verify the baseline coating process[44,45,47] and fabricate the baseline[48,49] fuel and three fuel "variants"[50,51,52,53] for the AGR-1 irradiation test has been completed. The small-coater work established (1) the feasibility of uninterrupted coating of the smaller UCO kernels, (2) the optimum small-coater batch size and fluidization conditions for the 350-µm UCO kernels, (3) small-coater process conditions for depositing an IPyC layer that has acceptable anisotropy and permeability and that closely approximates the density and microstructure of the German IPyC coating, and (4) small-coater process conditions for depositing a SiC layer of close to theoretical density that closely replicates the microstructure of the German SiC. In addition, the small-coater work included investigation of the use of argon dilution in SiC coating and investigation of a multiport gas distributor. Both of these activities were beyond the scope envisioned in the initial version of the program plan, but resulted in potential coating process improvements,[g] with the argon dilution option being tested as a fuel variant in AGR-1. The small coater work also provided data in support of the coating process modeling task. The baseline fuel process conditions selected by the program based on the results of the small coater work[16] are listed in Table 5.

In conjunction with the completion of baseline fuel coating using the process conditions identified in Reference 16, three variants on the baseline were defined with the primary objective of maximizing the prospects of successful performance of at least one of the fuels in AGR-1. Two of the variants were directed toward increasing the prospects of successful performance by providing diversity in the IPyC layer, while the third was directed toward diversity in the SiC layer. The following changes relative to baseline process conditions were made in producing the variant fuel types:[16]

- Variant 1–IPyC deposition temperature ~1,290°C
- Variant 2–IPyC coating gas fraction ~0.45
- Variant 3–SiC carrier gas = Ar + H_2; SiC coating temperature ~1,400°C.

The AGR-3/4 irradiation is primarily directed towards providing data on fission product transport from particles with failed coatings using driver-coated fuel particles in combination with DTF particles. The coating properties of the DTF particles are not a significant factor given that they are designed to fail early in these irradiations. So to meet schedule needs within allowed funding, the fuel for these irradiations was also produced in the laboratory-scale coater.[54,55,56]

g. Use of argon as a diluent in SiC coating allows the SiC deposition temperature to be reduced from 1,500 to about 1400°C, which reduces the potential for HCl attack of the UCO kernel during SiC coating. Coated particles fabricated with the multiport gas distributor had less faceting and less variability in coating thickness than coated particles fabricated with a single-inlet gas distributor.

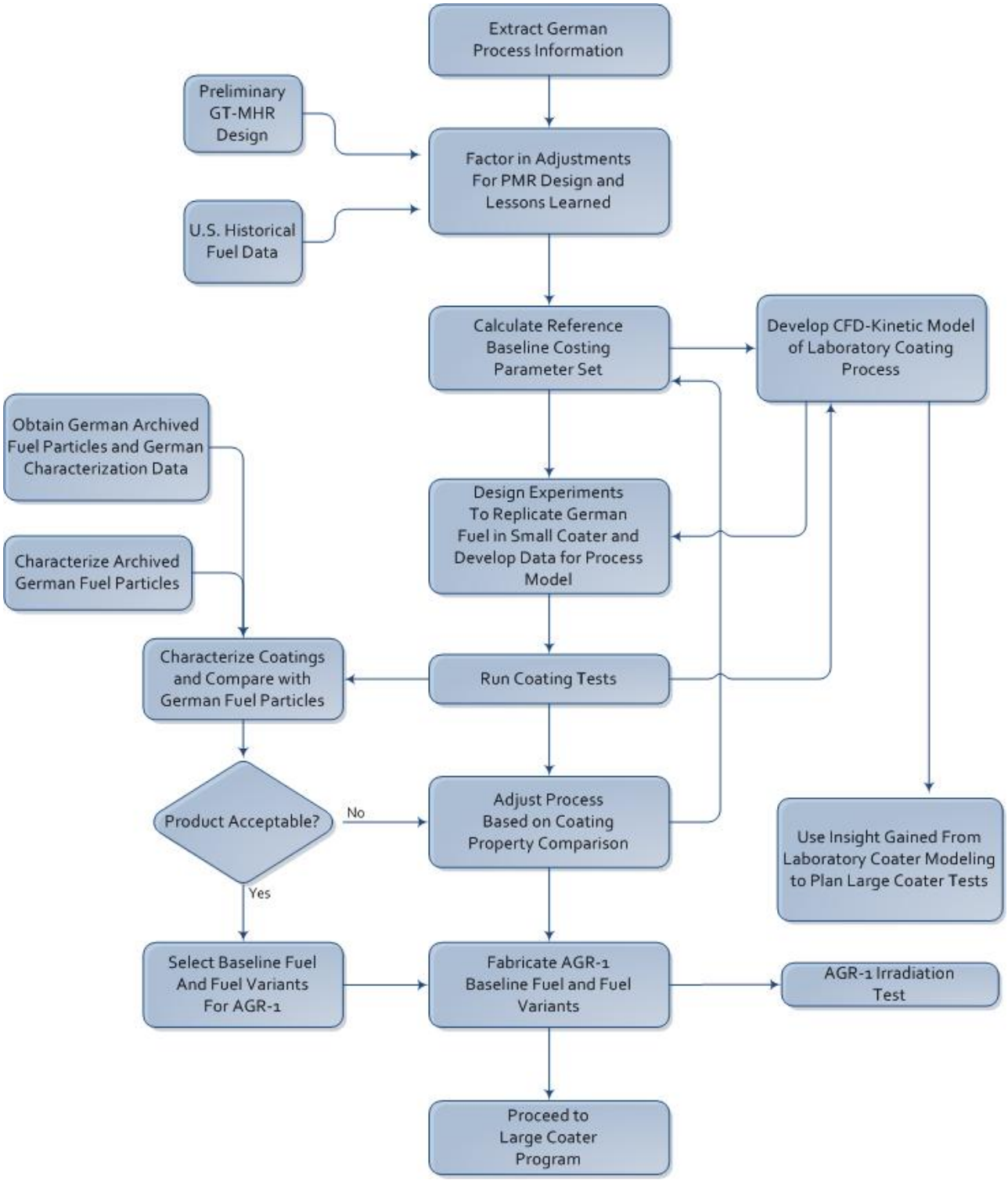

Figure 5. Logic diagram for small-coater work.

Table 5. Baseline coating process conditions for AGR-1.

Coating Layer	Coating Gas	Deposition Temp. (°C)	Coating Gas Fraction	Coating Rate (μm/min)
Buffer	C_2H_2	1,450	0.60	~20.0
IPyC	54% C_3H_6 46% C_2H_2	1,265	0.30	~3.0
SiC	MTSa	1,500	0.015	~0.25
OPyC	54% C_3H_6 46% C_2H_2	1,290	0.30	~3.0

a. MTS = Methyltrichlorosilane. Hydrogen was used as the carrier gas and diluent gas for the MTS.

The second phase of coating process development, which includes coating process development and fabrication of fuel particles for irradiation and safety testing in a prototypic production-scale coater, commenced in FY 2006 with the selection of a 6-inch coater[57] and development of a detailed plan for validation of the coater design.[58] Figure 6 illustrates the initial logic for the large-coater work. Following completion of coater design,[59] a coating furnace that was already installed and operational at B&W was modified/upgraded[60] and used for the large-scale coating work. To achieve AGR program goals of fuel qualification, coated particles for later experiments must be produced in a production-scale coater. The AGR program plan assumes that a 6-inch-diameter coater will be used to produce AGR-2 and later fuels. Multiple 6-inch-diameter coaters would be needed for a production facility, and economics may require ultimate demonstration and testing of fuel from a coater larger than 6-inches. However, the uncertainties of production facility requirements, the uncertainties inherent in scale-up, rough cost comparisons and programmatic budget, and schedule considerations led to the decision to develop, demonstrate, and use a 6-inch-diameter coater to produce TRISO-coated particles for AGR-2 and later fuels.

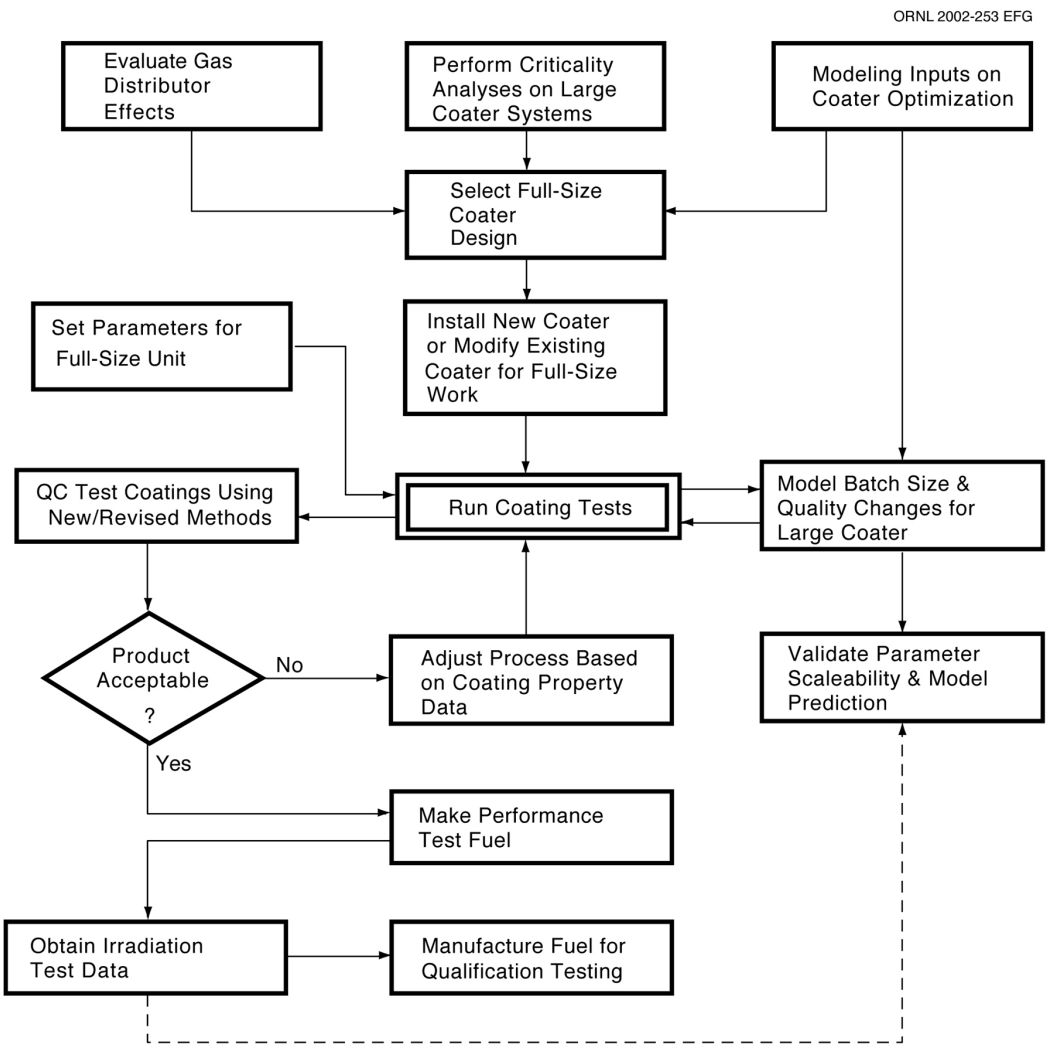

Figure 6. Logic diagram for large-coater work.

The initial coating runs in the 6-inch coater were made with surrogate and NUCO kernels to shake down the coater and adjust the reference set of coating conditions from small coater development to accommodate the characteristics the larger coater. The evidence that this objective was achieved was obtained by characterizing the coatings to verify that they have essentially the same desired properties as obtained in the laboratory-scale coater work.[61] As discussed in the following section, in-pile gas release results from AGR-1 show that all of the fuel types have performed well to fast fluence beyond values where particle failures were observed in earlier irradiations.[8] These results indicate that the baseline fuel and both pyrocarbon variants have successfully addressed the primary failure mechanism of excessive irradiation-induced pyrocarbon shrinkage. Thus, inclusion of variant particles, of either pyrocarbon or SiC variations, in AGR-2 was not considered necessary. A set of coating parameter variations similar to the small coater parametric studies was used to compare large coater characteristics and arrive at large coater process conditions that meet the AGR-2 fuel specification.

Coating process development at B&W using a 6-inch coater in FY 2009 culminated in the production and shipment to ORNL of four batches of UCO particles and one batch of UO_2 particles as candidates for use in AGR-2.[62] The pyrocarbon coating conditions were the same for all four batches. For two of the batches, a hydrogen-methyltrichlorosilane (MTS) mixture was used for SiC coating while for the other two, an argon-hydrogen-MTS mixture was used, similar to AGR-1 Variant 3. Characterization of compacts made from these particle batches revealed additional coating issues, namely uranium contamination and uranium dispersion in small populations of two of the batches of particles. Investigation confirmed that the uranium contamination was due to particle cracking subsequent to coating. Unloading equipment has since been modified to eliminate collisions that were likely the cause of these cracks. A third batch was rejected from use because of an anomaly that occurred during this coating run. Based on the characterization results, as well as the irradiation data from AGR-1 discussed in Section 5.2, UCO particles from the remaining UCO batch were selected for use in the AGR-2 UCO capsules. The UO_2 particles produced by B&W will be used in one AGR-2 capsule, with two additional UO_2 capsules containing particles produced by PBMR and AREVA.

Schedule considerations dictated that the fuel to be irradiated in AGR-2 be selected based on available coating experience and partial in-pile gaseous fission product release from the AGR-1 irradiation. Cesium release data and accident condition performance data obtained from PIE and post-irradiation heating of AGR-1 fuel will be evaluated to determine whether the metallic fission product retention and accident condition performance of the AGR-1 reference and variant fuels are acceptable and whether the performance varies significantly among the AGR-1 fuel types. These data, along with available in-pile performance data from AGR-2, will be used to determine whether any changes need to be made in the reference coating conditions used to fabricate the fuel particles for AGR-5/6 and AGR-7.

Fuel will be manufactured for the fuel qualification test AGR-5/6 with consideration of PIE and post-irradiation heating data for irradiated fuel from AGR-1, and in-pile gaseous fission product release data from the AGR-2 irradiation. Although the fuel manufacture effort will proceed with some financial risk to fabricate qualification test fuel with only fission gas release data from AGR-2, the risk is considered acceptable. The PIE data and heating test results from AGR-2 are expected to be available before beginning the fuel qualification irradiation test, AGR-5/6.

The coater charge size for AGR-2 particles was 1.3 kg of UCO kernels and 1.7 kg for the larger UO_2 kernels. Although the AGR-2 UCO charge size was larger than that for coating AGR-1 particles by a factor of 21, it was recognized that it was not optimal, and larger kernel charges would improve coater economics for a full-scale facility. Thus subsequent to AGR-2 particle production, tests were made of coater charge sizes up to 2.5 kg of 425-µm NUCO kernels. Based on limitations of coater auxiliary systems, a charge size of 2.0 kg of 425-µm NUCO kernels was selected as the maximum practical based on B&W's current coating system.

Coating results as of late 2009 led to several recommendations for continued development of the coating process to improve consistency in properties from run to run and to provide a basis for economical coating. These recommendations included both minor equipment upgrades and additional coating tests. A new pyrometer has been added to the coating furnace to improve understanding of coater temperatures. The hot sampler was redesigned. The new hot sampler was installed to take larger samples of buffer and IPyC-coated particles and to reduce the risk of loss of the sample cup into the particle bed during coating. The gas sweep MTS feed system was replaced with a turbovaporizer feed system to improve control of the MTS rate over the course of SiC coating. The effects of these changes, along with the increased charge size, were tested in FY 2010 and further testing will continue in FY 2011. Also, tests

of particle upgrading methods are being performed to better determine if tabling is required at all, and if so, what tabling parameters provide the optimal balance between yield and particle quality.

5.1.4 Compacting Process Development

Because of its apparent advantages, a thermosetting-matrix-based process was selected as the reference compacting process for the AGR Program.[63] In the selected process, the matrix, which had the same basic formulation (fractions of natural and synthetic graphite and binder) as the matrix used in the German program, is applied as an over-coating to the TRISO-coated particles. The over-coated particles are then poured into a mold and press molded to form the cylindrical fuel compact. No matrix other than the matrix on the overcoated particles is added to the mold. The compacts are heat treated at 900°C in argon to carbonize the matrix and then final heat- treated at about 1,800°C in a vacuum furnace. The equipment required for each of the compact fabrication steps (e.g., over-coating, molding, carbonization, and final heat treatment) was installed at ORNL and used to make high-quality compacts from surrogate and NUCO fuel particles. Formal qualification of the compacting process, with respect to AGR-1 specification requirements, was completed in 2005 and used to produce the baseline and variant fuel compacts irradiated in AGR-1.[64,65,66,67,68]

As previously noted, a fuel particle packing fraction of 35% was selected as the target for initial compact process development and for the AGR-1 irradiation test compacts. Actual achieved packing fractions exceeded the target, ranging from 36.5% for variant 3 to 37.9% for Variant 1. Following completion of AGR-1 compact fabrication, compacting development efforts focused on changes to process parameters to accommodate a larger particle (425 µm kernel) for AGR-2.[69] Increasing the matrix density was achieved, and the feasibility of higher packing fractions was investigated. Additional compacting process development was conducted to investigate different types of molding for forming the compacts (single action versus double action press, isostatic molding, vibratory molding, etc.), and to optimize the heat treatment processes. The results of these activities, documented in the 2008 ORNL report,[70] supported the compacting of fuel particles for the AGR-2 irradiation.

The ORNL scale compacting activities described above produced compacts for use in the AGR-1 and AGR-2 irradiations. While these efforts have successfully produced compacts for these irradiations, it has been recognized that adjustments to the process would be needed for scale-up to economical production of compacts in large quantities. A plan for developing a modified compacting process at B&W suitable for industrial scale production was established in 2009,[71] with the objective of using the scaled-up process for production of compacts for AGR-5/6 and subsequent irradiations. Goals of the compact scale up effort include:

1. Achieving a matrix density of ≥ 1.7 gm/cm^3 at low compacting forces
2. Obtaining the widest possible range for packing fraction adjustment without particle damage
3. Maximizing particle homogeneity within the compact
4. Eliminating the production of mixed waste
5. Maximizing process yields and minimizing scrap and recycle
6. Eliminating intermediate QC upgrading steps from the process
7. Utilizing standard industrial processes and equipment to the extent possible.

Natural and synthetic graphites make up 80% of the composition of the compacting matrix. Products from multiple suppliers of these source materials will be evaluated and qualified for use in compact

manufacture. Particle size, density, morphology, and impurity content are considered important in the selection of appropriate graphite materials for a production process.

Matrix powders have traditionally been manufactured by solvating the resin with an alcohol and then adding additional alcohol to make a mixture with a consistency of bread dough. This mixture is worked to ensure adequate mixing of the resin and graphite, then extruded into pencil like rods, dried, and crushed to a final desired agglomerate size. The ORNL process took this further, making the mixture into a slurry, mixing, drying, and crushing to a final agglomerate size. An alternative approach for preparing matrix has been demonstrated for the scaled-up process, that of jet milling. This approach takes the desired graphites with particle size up to and beyond 100 µm, mixes them with dry resin powder (typically less than 30 µm) then jet mills the mixture to achieve a consistent 6 to 8 µm particle size, dry powder. This change eliminates the use of a flammable solvent; eliminates multiple unit process steps; reduces process time from days to a few hours; produces the particle size needed for high density fluid bed over-coating, which is significantly smaller than desired for drum over-coating methods); and can use a variety of incoming natural flake graphite sizes and still achieve a very consistent final product.

A fluid bed over-coating approach was chosen for the scaled-up compact process because this approach eliminates the use of a flammable solvent which would result in mixed waste, ensures that the fuel particle is in the true center of the overcoated particle, is expected to allow a significant increase in the overcoat matrix density, and makes all over-coatings the same thickness without increasing the asphericity of the original particles[72]. This consistency eliminates the need for QC upgrading and allows charge determination on the basis of overcoated particle diameter, which can be directly correlated with fuel particle dimensions.

AGR-1 compact pressing was performed at room temperature and AGR-2 compacts were warm pressed (~65–75°C). Hot press compacting is planned for the scaled-up process. Hot press compacting may be distinguished from cold or warm pressing in that the pressing is done at a temperature at or above the Mettler drop point for the specific resin being used and the compact is held under pressure as the resin cures or at least partially cures in the tooling. For typical compression molding resins, hot pressing may be done from 110 to 170°C. The change to hot press compacting is being made because of the following:

- Melting of the resin to allow low pressure flow during compacting is considered the best way to achieve high matrix density with low compacting force, especially for high packing fraction compacts.
- Hot press compacting is expected to be the only method to allow successful compacting without matrix end caps. Because of the resin fluidity achieved during compacting, very low compacting forces are feasible. Without the low forces, pressing directly on overcoated particles would result in some particle damage.
- Hot press compacting is the standard method of manufacture for most compression molded parts in industry as it ensures die fill and ease of handling finished parts.
- Since curing occurs within the tooling, the finished compact has high strength and essentially no spring back during ejection (i.e., dimensional control of length and diameter is not altered during compact ejection from the die which otherwise results from relaxation of stresses). Dimensional consistency is ensured by controlling die diameter and operating temperature, so the only variable is in carbonization.

- The process is readily controlled and should allow mass production of compacts with no defects caused by the compacting process.

The final change proposed for the scaled-up process is to use a compression molding resin. Phenolic resins were originally developed for hot compression molding. Compression is used with elevated temperature to melt the resin and cause it to fill a die cavity, then set from a chemical reaction that would not allow it to re-soften, even with additional heating. A compression molding resin is considered necessary for compact scale up because of the following:

- Melting of the resin allows for low pressure flow during compacting and is thus considered the best way to achieve high matrix density with low compacting force, especially for high packing fraction compacts. Although the resin is only 20 wt% of the matrix, it is about 35% by volume and makes the matrix relatively fluid for a short time during compacting to achieve die fill.
- Curing the resin under pressure in the die results in a fully solid, strong, dimensionally stable compact at the green stage with essentially no springback during compact ejection from the die. Therefore, the fuel particles are fully protected from damage even before removal of the compact from the tooling. This is especially important for high packing fractions because as the packing fraction increases there more and more fuel particles may touch the exterior of the compact.

The initial choices for resins are standard novolac compression molding resins that pass their Mettler drop point at about 110°C and then cure in ~90 seconds at 140°C. As curing is through chemical reactions, the curing time is a direct function of temperature, so it would be ~23 seconds at 160°C or ~720 seconds at 110°C. High purity versions of these standard resins available that are being tested, but currently do not have a basis to determine if they are necessary. The following steps will be taken to demonstrate the scaled-up compact fabrication process:

1. Fabricate, characterize and compare properties of matrix-only compacts using both AGR-2 and scaled-up process methods. [*Completed*]
2. Fabricate, characterize and compare properties of compacts made with both AGR-2 process and the scaled-up process.
3. Complete resin and graphite selection.
4. Complete over-coating tests larger overcoater (i.e., Vector).
5. Determine the range of packing fractions both processes are capable of achieving.
6. Determine strength of matrix-OPyC bond in compacts made using both AGR-2 and scaled-up methods.
7. Complete final design of B&W compacting line. [*Completed*]
8. Procure and install compact line at B&W. [*Underway*]
9. Perform tests of B&W compact line with surrogate particles to determine compacting parameters.
10. Perform tests with NUCO particles to finalize parameters for qualification tests.
11. Perform process qualification tests.

B&W has completed the final design for the AGR Pilot Scale Compacting Process.[73] The estimated production rate for a production-scale compact fabrication line is 3 million compacts per year or 12 compacts per minute. A throughput rate requirement was not imposed on the pilot scale compact process

except that it should be capable of demonstrating how the process would be operated in a production environment and provide a product that can be shown to be identical to a production-line product from the technical and regulatory viewpoints. The design report contains process flow diagrams, a discussion of each section of the process, mass balances, an equipment layout diagram, siting and facility modifications, staffing requirements, cost estimates, nuclear criticality safety considerations, industrial health and safety considerations, nuclear materials control and accountability considerations and a radiation protection analysis.

B&W performed a "make/buy" evaluation for the matrix material and concluded that it was advantageous to subcontract matrix preparation rather than include equipment to produce the matrix as part of the compact process.

B&W obtained bid packages from multiple vendors for all major equipment items, including the overcoater, compact press and furnace. Orders for all major equipment items were placed by May 31, 2010 with some equipment delivered in the fourth quarter of FY 2010 and the remainder will be delivered in the first quarter of FY 2011.

5.1.5 QC Methods Development and Application

For the AGR-1 fuel, B&W focused mainly on characterization of kernels while ORNL addressed characterization of coated particles and compacts. Both organizations developed procedures in the other areas to support internal activities as well as provide a basis for cross checking results produced in the other organization.

5.1.5.1 Oak Ridge National Laboratory

The planned QC methods development included tasks to (1) set up a QC laboratory containing the equipment needed for basic QC inspection of coated particles and compacts, and where appropriate, (2) develop new methods needed for enhanced characterization of kernels and coatings. The first of these tasks has been completed by setting up a dedicated coated-particle fuel characterization laboratory at ORNL. This laboratory provides the following capabilities for inspection and testing of kernels, coated particles, and compacts to demonstrate compliance with fuel product specifications:

- Ceramography to provide images for analysis
- Automated image analysis for kernel and particle diameter, sphericity, and coating thickness measurements
- Density gradient column for PyC and SiC sink-float density measurements
- Mercury porosimetry for measuring kernel, buffer, and coated particle envelope density and for PyC surface connected porosity measurements
- A new technique for measuring PyC coating anisotropy[74] (e.g., Two Modulator Generalized Ellipsometry Microscope [2-MGEM])
- Compact measurements including length, diameter, mean uranium loading, total mass, and defective IPyC and OPyC coating fractions
- Leach-burn-leach testing of fuel compacts to determine the defective SiC and uranium contamination fractions, and the quantity of specified impurities outside the SiC layer.

- Visual examination of burned-back particles to determine "gold spot" defect fraction and inspection of ceramographically mounted particles for SiC inclusions. This latter method is required as a substitute for the gold spot determination for particles with fine grained SiC microstructure.

In addition to the above capabilities, test methods using other analytical capabilities have been developed at ORNL for use on the program in support of coating process development. These capabilities include:

- Scanning electron microscopy
- Transmission electron microscopy
- X-ray diffractometry
- Chlorine leaching/x-radiography to characterize the permeability of IPyC coatings
- Uranium content analysis
- Strength testing of PyC[75] and SiC layers[76]
- Characterization of ZrC-coated particles.[77]

ORNL has also procured a high-resolution x-ray tomography system from Xradia Corporation. This system, which has been optimized for nondestructive examination of coated-particle fuel, is capable of producing 3-D tomographic images of fuel particles with an unprecedented resolution of 1.6 µm. It is expected that the 3-D images obtained with this system will contribute to a better understanding of the relationships between coating microstructure and fuel performance.

The characterization methods established at ORNL were formalized in work instructions that were used for QC inspection and testing of fuel fabricated for AGR-1 and AGR-2. Quality assurance procedures have also been implemented to control these work instructions, identify and control inspection/test records, train and qualify inspection personnel, and calibrate and ensure the in-calibration status of measurement and test equipment. These methods and procedures have been used to characterize the four fuel types irradiated in AGR-1[49,51,52,53] and demonstrate compliance with the AGR-1 fuel specification[16] in accordance with NQA-1 requirements. The methods and procedures were subsequently adjusted where necessary and applied to characterization of the four fuel types being irradiated in AGR-2.[78,79,80,81,82] Additional methods were established to support SiC fracture strength measurements.[83]

ORNL has supported the transfer of coated-particle fuel characterization technology to B&W and has applied the QC capabilities developed for the fabrication and characterization of AGR-1 fuel in support of the development of large coater fabrication capabilities at B&W for the AGR-2 fuel. In some cases (e.g., buffer density, SiC defects), ORNL has duplicated characterization work performed by B&W to enhance confidence in the data and demonstrate consistency with the measurements performed on AGR-1 fuel. In other cases (e.g., pyrocarbon anisotropy, heavy metal contamination, SiC defect fractions), ORNL has provided characterization data that were used to demonstrate compliance with the AGR-2 fuel specification.

ORNL has also supported an IAEA Coordinated Research Project on HTGR Fuel Technology by providing coated-particle specimens and characterizing particle coatings on specimens from the U.S. and other participating countries in a particle characterization round robin exercise.[84] The results of this activity will enhance confidence in characterization data among the participating countries and, in the case of the 2-MGEM measurements, a correlation between results using this advanced method and

traditional anisotropy measurements. This will strengthen the basis for interpretation of historical fuel performance data in establishing specification limits applicable to the 2-MGEM method.

5.1.5.2 Babcock and Wilcox

B&W has implemented QC processes and analytical procedures to ensure that the kernels and coated particles they produce will comply with AGR specifications. For kernels, metallographic, chemical and radiological procedures were in place to determine kernel diameter, kernel aspect ratio, the chemistry of kernel batches and composites (carbon, oxygen, uranium, and 15 impurities), ^{235}U enrichment, and envelope density. The AGR-1 fuel specification contained C/U limits and, in lieu of developing a method to analyze the chemistry of individual kernels, an image analysis method was used to provide a basis for acceptance of the AGR-1 kernel lot relative to this specification.[85,86] This method was successful in demonstrating that AGR-1 kernels were acceptable because AGR-1 kernels were predominantly two phases: UO_2 and UC. However, the method breaks down for UO_2-UC-UC_2 kernels. Process improvements have resulted in kernel lots with much more homogeneous chemistry than seen in AGR-1, as confirmed by photographs from 8 lots (>110 kg) of kernels produced after AGR-1.

Particle characterization capabilities were reviewed before starting prior coating tests using the 6-inch-diameter coater at B&W,[87] and capabilities for all required analyses except pyrocarbon anisotropy were in place prior to AGR-2 particle coating. In 2010, B&W placed an order for a 2-MGEM to determine pyrocarbon anisotropy. This instrument will be installed at B&W in FY 2011.

As part of scale up of the compact process, B&W is procuring all instruments needed to perform required quality assurance analyses on compacts and on particles deconsolidated from compacts.

As particle defect fraction specifications have been reduced from AGR-1 to AGR-2 and will be again for AGR-5/6 fuel, larger and larger quantities of compacts are destroyed in time-consuming deconsolidation/ leach-burn-leach (LBL) analyses. Continued development of fuel characterization techniques is needed, driven by cost considerations for full-scale fuel manufacturing. An initial study to estimate fuel costs will include characterization costs in order to guide future development of characterization methods needed for industrial fuel manufacturing.

5.1.6 Fuel Product and Process Documentation

Documents have been produced in compliance with QA requirements in the following areas:

- Fuel specifications
  - AGR-1 fuel[16]
  - AGR-2 fuel[33]
  - AGR-3 and 4 fuels[34]
  - Preliminary NGNP fuel[30]
- Kernel process development and kernel production
  - Chemistry improvement[38,39]
  - Process development[40]
  - NUCO kernel production[35]
  - AGR-1 LEU UCO kernel production[36,85,86]

- AGR-3 and 4 DTF LEU UCO kernel production[37]
- AGR-2 LEU UCO kernel production[88]
- AGR-2 LEU UO_2 kernel production[89]

- Coating process development and coated particle production
  - Characterization of German particles[44]
  - Characterization of HRB-21 particles[45]
  - Coating of depleted UO_2 kernels[47]
  - IPyC deposition study[43]
  - Fabrication of baseline and variant particles for AGR-1[48,49,50,51,52,53]
  - Coater scale-up plan[57]
  - Test plan for validation of 6-inch coater design[58]
  - Coating process modeling[46]
  - 6-inch coater and control system upgrade[60]
  - AGR-2 particle characterization[90,91,92,93,94,95,96, 97,98]
- Compacting process development and compact production
  - Compacting development plan[63]
  - Compacting Scale-up mass balance and process flow evaluation[99]
  - Fabrication of baseline and variant compacts for AGR-1[64]
  - Process development for particles with 425 µm kernels[69]
  - Data compilations for AGR-2 compacts[100,101, 102,103,104]
- QC methods development and application
  - Anisotropy measurement method[74]
  - Assessment of B&W capabilities[87]
  - Data compilations for AGR-1 baseline and variant particles[49,51,52,53]
  - Data compilation for AGR-3 and 4 driver fuel particles[54]
  - Data compilations for AGR-1 baseline and variant compacts[65,66,67,68]
  - Data compilations for AGR-2 kernels, particles, and compacts.[20]

5.1.7 Technology to Mass Produce Particle Fuel

Efforts to date have focused on establishing fabrication and QC capability, and on fabrication of fuel to support the AGR-1, AGR-2, and AGR-3/4 irradiations. Coater equipment and process development was conducted for scale-up from the small coater used for AGR-1 and AGR-3/4 fuels to the 6-inch coater used for AGR-2 fuel. Although additional scale-up may be required for large scale commercial fuel production, the 6-inch coater supports production of fuel for subsequent AGR fuel program testing and

provides sufficient capability (multiple coaters operating in parallel) to produce a first core and initial reloads. Also, an effort is under way to scale-up the compacting process to support economical mass production of compacts.[71] The scaled-up process is planned to be used to produce compacts for the AGR-5/6 and subsequent irradiations.

As of mid-2009, the capacity of the B&W kernel fabrication process was 136 kg UCO kernels per year (122 kg as U), based on a single shift, 5 day per week, 40 week per year operation. The primary bottleneck limiting this rate is the kernel drying station in the forming section of the process. Adding a new second drying station would allow running the forming system 4 days per week, doubling the expected throughput to about 274 kg kernels/year (245 kg as U). Assuming the full-scale Fuel Fabrication Facility is designed for a fuel replacement rate of 2,270 kg U/yr, the B&W kernel fabrication line at twice the current capacity would demonstrate a rate of only 9% of that which would be required in the full scale facility. Changes that should be considered for producing kernels economically at a full-scale rate include larger diameters ADUN dissolvers, redesigned ADUN filtration to increase efficiency, using a rolling heel in ADUN dissolvers, redesigning the kernel forming column for continuous operation, eliminating manual transfers in kernel forming, use of larger wash/dry vessels and/or an alternative kernel drying system, and evaluation of alternative, less labor-intensive upgrading methods.

Improvements have been made to the kernel fabrication process in the course of producing NUCO kernels for coating tests. The yield of product from the forming process increased from 86% for AGR-1 kernels to 95% for FY 2010 NUCO kernels and from 73% to 88% for the sintering process. The kernel sintering furnace batch size has been increased from 750 g for AGR-1 kernels to 1,700 g for AGR-2 kernels to 3,300 g for FY 2009 NUCO kernels. In FY 2010, modifications to ADUN preparation have been made. An evaluation of an alternative kernel upgrading method will be performed in FY 2011. Recommendations for additional short-term modifications include:

1. Modify the ADUN filtering system to reduce maintenance and reduce scrap loss
2. Review the ADUN preparation procedure with the goal to reduce cycle time from 4–5 days to 3 days
3. Add drying station to double throughput. The new drying station would have improvements from the existing one including temperature control for each product collector and data logging of inlet and outlet temperatures.

AGR-2 particles were coated in a 6-inch diameter coater using a charge size of 1.3 kg UCO kernels. Tests of charge sizes up to 2.5 kg were made in late 2009 and early 2010, and while feasibility at this increased charge size was demonstrated, it was found that current B&W coater support systems will likely limit the charge size to about 2 kg. At this charge size, about 1,570 coating runs per year would be required to support a fuel replacement rate of 2,270 kg U/yr. Prior to scale up of coating for AGR-2 fuel, an economic evaluation of coater diameter was performed.[105] This study left open the possibility of reevaluating coater diameter after AGR-2. As fuel requirements for the NGNP become more definite, a re-evaluation of coater diameter is needed to ensure adequate coating data will be obtained to support the design of the full scale facility.

Particle upgrading (sieving or tabling out of round, over-sized material) is a slow and labor intensive step. Studies of particle upgrading were performed in FY 2010 and will complete in FY 2011; continued studies should consider elimination of the tabling step and possibly, in conjunction with studies minimizing particle defects, evaluate alternative upgrading methods.

AGR-1 and 2 compacts were produced at ORNL in small-scale equipment. A new process has been designed and new equipment procured for fabrication of AGR-5/6 compacts at B&W. The equipment was

designed to be representative of the type of equipment expected to be used in a large-scale operation.[h] To achieve the scale-up objectives, some changes were needed from methods and equipment used for AGR-1 and 2 compacts.

5.2 Fuel and Materials Irradiation

The tasks required for each irradiation and the irradiation tests to be conducted during the AGR fuel program are described in Sections 3.2.2 and 3.2.3.

5.2.1 AGR-1 Irradiation

The status of tasks for the first irradiation, identified as AGR-1, is as follows:

1. *Test Specification*. A preliminary AGR-1 irradiation test specification issued by INL in September 2003[106] provided initial guidance for planning and test train conceptual design. This was superseded in March 2004 by Revision 0 of the test specification, updated by Revision 1 in July 2004,[107] which has served to guide the final design of the AGR-1 test train.
2. *Test Train and Supporting Systems Technical and Functional Requirements*. Technical and functional requirements for both the test train and external systems were issued by INL as Revision 0 in July 2004. Revision 1 of the requirements for the test train was issued in December 2004,[108] and for external systems in March 2005.[109] These documents incorporated AGR-1 test specification requirements and provided more detailed guidance for development of the AGR-1 test train design.
3. *Test Train and Supporting Systems Design*. The conceptual, preliminary, and final designs were completed, and the corresponding design reviews were held in September 2003, August 2004, and March 2005,[110] respectively. Supporting analyses included neutronics, thermal and stress/seismic evaluations[111] and projections of irradiation conditions and fuel performance.[112] Additional analysis and design refinements (e.g., thermocouple type and placement[113]) were conducted following the final design review to resolve outstanding issues.
4. *Test Train and Supporting Systems Fabrication/Assembly*. Long lead items for the test train and supporting systems were ordered in FY 2005 to support the schedule for insertion in ATR. The test train was complete and ready for insertion as scheduled at the end of FY 2006.[114]
5. *Approval of Test Articles*. The baseline and variant fuel compacts to be irradiated in AGR-1 were approved by INL based on a review of the particle and compact characterization data.[115]
6. *Review/Approval of Test Train Final Design and Fabrication.* A review of the AGR-1 test train from the standpoint of experiment safety was conducted and documented in advance of test train insertion in the ATR at the end of FY 2006.[116]
7. *Irradiation*. Prior to the irradiation, a document summarizing the objectives and success criteria of the experiment was generated,[117] a test plan was produced to guide the operation of the test train,[118] and a pretest prediction provided projections of burnup and fast neutron fluence accrual, fuel temperature distributions, particle failure fractions, and Kr-85m R/B ratios in each capsule of the test train.[119] The pretest prediction of no particle failures in any of the AGR-1 capsules and the predicted R/B ratios have been supported by data from the irradiation.[120,121] Actual initial R/B ratios were approximately an order of magnitude lower than predictions, likely because the pretest prediction assumed the matrix heavy metal contamination had the same enrichment as the kernels (19.8%), while the

[h] S. Niedzialek, AGR Compacting Pilot Facility Final Design Report, BWED 10-144, Revision 0, May 18, 2010.

behavior of the R/B during the irradiation indicated that the contamination was predominately natural uranium. Actual R/B values increased significantly during the course of the irradiation, as would be expected for natural uranium contamination, caused by the buildup of plutonium isotopes as the irradiation proceeds.

5.2.2 AGR-2 Irradiation

The status of tasks is as follows:

1. *Test Specification*. The irradiation test specification for AGR-2 was issued in May 2009,[122] and was updated to Revision 1 in June 2010.
2. *Test Train and Supporting Systems Technical and Functional Requirements*. The technical and functional requirements for the external systems (e.g., temperature control and fission product monitoring) written for AGR-1 were utilized for AGR-2 because the same systems were used without modification. The technical and functional requirements for the test train were issued by INL as Revision 0 in May 2009, were superseded by Revision 1 in early April 2010, and updated to Revision 2 in later April 2010. These documents incorporated AGR-1 test specification requirements and provided more detailed guidance for development of the AGR-1 test train design.
3. *Test Train and Supporting Systems Design*. The preliminary and final designs were completed, and the corresponding design reviews were held in October 2008 and May 2009, respectively. Supporting analyses included neutronics, thermal and stress/seismic evaluations and projections of irradiation conditions and fuel performance. A design change review was conducted in April 2010 to address changes in the design that occurred after the final design review including the high (operating) temperature capsule.
4. *Test Train and Supporting Systems Fabrication/Assembly*. Long lead items for the test train and the external system maintenance items (e.g., mass flow controller calibrations, fission product computer updates, etc.) were ordered in FY 2009 to support the schedule for insertion of AGR-2 in mid-2010. The test train was complete and ready for insertion as scheduled by early June (ahead of the June 30 milestone) and inserted in the reactor in mid June 2010.
5. *Approval of Test Articles*. The UCO and UO_2 fuel compacts to be irradiated in AGR-2 were approved by INL based on a review of the particle and compact characterization data.
6. *Review/Approval of Test Train Final Design and Fabrication*. A review of the AGR-2 experiment from the standpoint of reactor and experiment safety was conducted and documented in advance of test train insertion in the ATR in June 2010.
7. *Irradiation*. The irradiation began June 23, 2010, and is proceeding.

5.2.3 AGR-3/4 Irradiation

The status of tasks is as follows:

1. *Compact Specification*. Compact specification has been prepared and issued.
2. *Test Train and Supporting Systems Technical and Functional Requirements*. Technical and functional requirements for both the test train, flux trap support hardware and the external systems (e.g., new temperature control and fission product monitoring systems) were drafted by INL in 2009 and utilized to support the preliminary design of these items in 2010.

3. *Test Train and Supporting Systems Design*. The preliminary design of the test train and supporting systems was initiated in 2009 and the preliminary design reviews for the test train, flux trap hardware and the supporting systems were completed in August 2010.

5.3 Safety Testing and PIE

A PIE Working Group with participants representing relevant organizations was formed to prioritize and guide preparations for the AGR-1 PIE and safety testing. The following efforts were completed in the early stages of the AGR-1 PIE and safety testing planning:

- *Identification of Existing Capability.* Assessments of existing capability for conduct of PIE and safety testing were performed and the results were documented in reports addressing capabilities at INL[123] and ORNL.[124]
- *AGR-1 PIE and Safety Testing Requirements.* A preliminary set of requirements was produced prior to the beginning of the AGR-1 irradiation[125] as a reference for planning the establishment of the needed capability. The requirements included allowances for different scenarios depending on the fuel performance observed during the AGR-1 irradiation.
- *PIE and Safety Test Facility Development Plan*. [126] A plan for facility development based on the preliminary requirements was produced to guide the efforts needed to be ready to proceed when the AGR-1 test train was available for disassembly.
- *AGR-1 Post-irradiation Examination Plan*. [127] This plan includes a detailed description of the specific activities planned for the AGR-1 experiment as well as prioritization approach for analyzing the different fuel variants contained in the six capsules.

Efforts to upgrade existing facilities and install new facilities as necessary to execute the required scope in a timely manner have been conducted in accordance with the PIE plan for the AGR-1 test train.[128] This includes developing additional capacity for high-temperature accident release testing, preparing hardware and methodology for inspection, disassembly, and measurement of the AGR-1 and subsequent test trains, upgrading sample preparation and microanalytical capabilities (ceramography, electron probe microanalyses), upgrading and developing new irradiated particle handling and inspection instrumentation, upgrading and developing new irradiated microsphere gamma analysis instrumentation, and modifying and installing compact deconsolidation and LBL equipment in shielded hot cells. The work has also included development of new procedures to govern PIE activities using this specialized equipment.

The AGR-1 test train was shipped from the ATR Complex to the Materials and Fuels Complex on March 25, 2010, and post-PIE activities are currently in progress in accordance with the AGR-1 PIE Plan, including capsule disassembly and metrology, and compact and test train gamma scanning. One shipment of compacts was completed from INL to ORNL in FY 2010 to initiate PIE activities there as well. Three additional shipments of AGR-1 compacts will be completed in FY 2011. A summary of PIE and safety testing results from the AGR-1 experiment will be included in future revisions to this plan.

5.4 Fuel Performance Modeling

An initial version of the theory and model basis for the PARFUME code has been published.[129] More detailed discussions will be provided in future versions of this plan.

To date, the following activities have been performed to enhance or apply the PARFUME code as part of the AGR program and an I-NERI with France.[130]

Normal Condition Modeling

- A consistent set of material properties has been incorporated, including unique correlations for PyC creep and bond strength between layers for U.S. and European fuel.
- Variable irradiation condition history (burnup, fast neutron fluence, and temperature can each independently vary as a function of time) has been incorporated.
- Various fuel geometries have been modeled, including single particles, spherical, and cylindrical compacts.
- Failure probabilities have been calculated based on Monte Carlo or multiple direct integration techniques.
- Several possible failure modes have been evaluated, including: pressure vessel failure, stress failure due to asphericity, irradiation-induced shrinkage and associated cracking of the IPyC, partial debonding of IPyC from SiC, kernel migration, and Pd attack.
- An iterative thermal model has been developed that accounts for temperature-dependent material properties and dimensional change (shrinkage, swelling, and gap formations) of the fuel components.
- Fission product gas release correlations have been incorporated that account for direct fission recoil from the kernel to the buffer and diffusive release resulting from intrinsic, thermal, and radiation-enhanced diffusion.
- A CO production model has been developed that considers burnup, temperature, uranium enrichment, and fuel composition in the correlations.
- Internal particle pressures have been calculated according to the Redlich-Kwong equation of state.[131]
- R/B-rate[i] ratios for selected short-lived isotopes from failed particles and uranium contamination have been calculated.

Accident Condition Modeling

- Failure probabilities have been calculated for an accident heat-up condition that follows a given irradiation history.
- Models describing fission product transport under accident conditions have been developed.[132]
- Models accounting for thermal decomposition of the SiC layer and chemical interactions between fission products and the SiC have been completed.

Modeling Results

- Benchmark activities under the IAEA Coordinated Research Program (CRP)-6 on coated-particle fuel technology have been undertaken. Benchmark cases include simple analytical problems, single particles with increasing complexity, and past and planned irradiation experiments, and past and planned fuel safety tests. Calculations of normal condition cases[133] and for accident condition cases[134] have been completed.

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i. R/B is a measure of the fractional release rate of short-lived fission gases and has been traditionally used as an indicator of fuel performance.

- A status report on incorporating ZrC modeling within PARFUME has been published.[135] Sample analyses indicate that failure probabilities for fuel particles with ZrC are high based on current strength data for ZrC, which appear to be too low. New strength data for ZrC to be forthcoming from a joint Japan-ORNL I-NERI research program should improve the PARFUME calculation results.
- Scoping calculations of fuel failure probability response functions for UCO PMR and UO_2 PBR fuel were performed as functions of fluence, burnup, and operating temperature under normal irradiation conditions.[136] The PARFUME calculations used some conservatism in the assumptions of key TRISO material properties. Irradiation temperatures of UCO fuel at high fluence and burnup below 1,000°C and above 1,400°C produced significant fuel failure probabilities. For UO_2 fuel, irradiation temperatures above 1,000°C produced significant failure probabilities at high fluence and burnup. For these calculations, high fluence is defined as 3.5×10^{25} n/m^2 and high burnup is defined as 12% FIMA.
- Pretest evaluation of the AGR-1[78] and AGR-2[137] experiments have been completed.

5.5 Fission Product Transport and Source Term

The scope and schedule of the research plan in the area of fission product transport has been reviewed[138] and the flow of experimental data from its acquisition to its end use has been delineated. Special requirements for data acquisition were noted, and recommendations were offered and have been incorporated into this revision of the plan.

The HFR-B1 experiment was reviewed[139] and a report published summarizing the results, recommendations for further data analysis, and lessons learned which are to be applied to the design of the AGR-3/4 experiments. Fuel hydrolysis data from Capsule 3 have been extensively analyzed and used for model development. However, fission gas release data from Capsules 1 and 2 have not been quantitatively assessed. These data were evaluated and judged to have substantial value for development of improved fission gas release models. Consequently, it was recommended that fission gas release data from Capsule 1 (burnup effects) and Capsule 2 (temperature effects) should be further analyzed, including a quantitative comparison of calculated gas release with the measured data, and the results used to upgrade the models. In order to characterize fission product transport in core materials, it was determined that a one-dimensional test geometry is needed along with a well-defined fission product source term (DTF particles serve that purpose well). It was also determined that a complete mass balance is required for the fission products of interest measured in the post-PIE of each irradiated capsule.

Review of the HFR-B1 Capsule 1 database was completed.[140] The review concluded that the amount of quantitative PIE data available from HFR-B1 Capsule 1 is insufficient to validate the design methods for predicting fission metal release during normal operation. The primary reasons for this conclusion are twofold: (1) the time lapse between completion of the irradiation phase of the experiment and the PIE caused 250-day Ag-110m to decay below detection limits, and (2) the reported gamma scan results for most of the Capsule 1 components are relative count rates that were not reduced to absolute radionuclide inventories (i.e., the requisite calibration measurements were not made).

The AGR-3/4 experiments were redefined[141] to be irradiation experiments with identical one-dimensional geometry for measuring fission metal transport in compact matrix material and fuel element graphite. A single stack of fuel compacts containing a known fraction of DTF particles will be surrounded by three annular rings: (1) an annulus of fuel-compact matrix material; (2) an annulus of fuel element graphite; and (3) an annulus of char-impregnated graphite to act as a sink for fission products. It was also recognized that measurements of fission product sorptivities in fuel-compact matrix, core

graphite, and primary circuit alloys would need to be measured in out-of-pile experiments. A test specification for these measurements was published in 2006.[142]

The AGR-3 and AGR-4 experiments have been combined in one experiment, AGR-3/4, to be conducted in the northeast flux trap of the ATR, and is currently in design. As conceived, the test train will contain 12 separate capsules, each independently controlled and monitored for fission gas release, and each containing four 1 inch long compacts. Two grades of fuel element graphite will be tested, GrafTech PCEA and Toyo Tanso IG-110. Target compact operating temperatures are 900, 1,100, 1,250, and 1,400ºC. Target fast fluence ranges from about 2.5×10^{25} to about 4×10^{25} n/m^2 and target burnup ranges from approximately 12 to 20% FIMA. The DTF fraction is specified to be 1×10^{-2}. It has been decided not to include piggyback capsules of unbonded fuel particles in AGR-3/4 to minimize thermal discontinuities in the experiment. A judgment was made that measurements of the diffusivity of metal fission products in graphite materials was a higher priority than the measurement of the diffusivity of gaseous fission products in PyC layers from irradiated, unbonded BISO particles.

Analysis of plateout data from the COMEDIE SR-1 test was completed using the PADLOC code.[143] The evaluation of the SR-1 silver and cesium plateout data with the PADLOC code, using the GA material property data as input, did not support the traditional design goal of demonstrating the code predictions are accurate to within a factor of 10. These uncertainties result from the very large uncertainties in the material property data, especially the sorption isotherms, used as input to the models and the large uncertainties in the transport models, themselves. The GA oxidized silver isotherm over predicts the Ag-110m concentrations on the heat exchangers in SR-1 by 1 to 2 orders of magnitude. The GA-oxidized cesium sorption predicts a small sorption limitation at the entrance of the SR-1 heat exchanger that is not evident in the experimental data. The poor correlation between experimental data and code calculations for both silver and cesium (especially in the mass transfer region) especially at the high temperatures, implies that some part of the deposition process is not being modeled correctly by PADLOC.

A test plan for single-effects fission product plateout and liftoff tests has been published.[144] An experimental program is proposed comprising three types of tests: laboratory-scale sorption measurements, out-of-pile loop tests, and decontamination tests. The deposition behavior of four radionuclides (Ag, Cs, Te, and I) on three different metals will be determined as a function of temperature, partial pressure, and coolant chemistry (surface oxidation state). The number of sorption measurements (equilibrium surface concentration as a function of temperature and partial pressure) needed to derive reliable isotherms is large. Relying exclusively on loop tests to produce all the required data would be too expensive and time consuming; consequently, much of these data will be generated in simpler laboratory-scale test facilities. The out-of-pile loop tests are necessary to investigate the effects of flow and high system pressure on deposition and to perform in situ liftoff tests. The decontamination tests are designed to investigate methods for decontaminating primary circuit alloys to reduce dose rates during reactor maintenance. It is proposed to investigate various decontamination protocols, including (a) conventional mechanical methods (e.g., high-pressure water and/or steam), (b) electrochemical leaching to chemically remove surface layers, and (c) commercial decon protocols developed for LWRs (e.g., low oxidation-state metal ion). Tentative choices of the three alloys to be used in the sorption, loop, and decontamination tests, subject to change as the design of the NGNP becomes better defined, are IN 100 (turbine blades), Inconel 617 (hot duct, intermediate heat exchanger), and 316H SS (recuperator). To provide a context in which to consider the proposed test program, the design methods for predicting plateout and liftoff are summarized, and the experience with such test programs is described in some

detail. The estimated cost and schedule for the proposed program are arranged in a work breakdown structure consistent with the AGR technical program plan.

Requirements for an in-pile fission product transport loop have been published.[145] The loop should be representative of the primary circuit of a HTGR and should be capable of being blown down in situ through a depressurization train, which collects quantitatively the condensable radionuclides re-entrained and carried out of the loop. The loop should have facilities for the injection of pre-characterized particulate matter (dust) and for the injection of steam and/or liquid water. The functional and technical requirements for such a loop are defined, including (a) physical configuration, (b) preoperational characterization, (c) operational characteristics during normal operation and depressurization transients, (d) PIE, and (e) quality assurance. Currently, no operational loop facilities in the world are capable of meeting these requirements; consequently, a new test facility will need to be designed and constructed or an existing facility (e.g., the PG-1 loop at NIIAR, Demitrovgrad, RF) will need to be extensively refurbished. Previous experience with in-pile fission product transport loops has been reviewed by General Atomics. As stated earlier the NGNP project continues to evaluate the need for an in-pile loop test given cost and schedule considerations. Subsequent evaluation indicated that a mockup of the reactor should be included in the test facility as discussed below.

A review of the understanding of radionuclide transport in the VLPC has been completed[146] that includes an evaluation of the applicability of data and design methods from studies of fission product behavior in LWR containments. Conclusions and recommendations from the VLPC review include:

1. The presence of dust in the primary circuit can increase the fractional re-entrainment of plateout activity during rapid depressurization accidents and, thus, the quantities of radionuclides released into the VLPC.
2. Fission products have such low mass concentrations in the VLPC that compound formation between radionuclides such as CsI is not expected and condensation into aerosols is also not expected, that is, fission products will be present in the VLPC in elemental form.
3. Iodine will chemisorb on surfaces.
4. Water-reactor data cannot be used to refine or validate the design methods for predicting radionuclide transport in the VLPC because the mass concentrations, chemical speciation, and physical forms are markedly different.
5. A number of the design methods developed for water-reactor containment analysis may be used for the analysis of radionuclide transport in the VLPC, especially system codes such as MELCOR and MACCS, providing modular gas-cooled reactor specific input data are available.
6. New DDNs need to be defined for characterizing radionuclide transport in the VLPC for the accident scenarios postulated for the NGNP.
7. A test plan needs to be developed defining an experimental program to satisfy the DDNs.
8. The particulate matter in the Japanese High Temperature Test Reactor should be characterized, because it would provide the best near-term definition of the dust in a prismatic block modular gas-cooled reactor. JAEA should be encouraged to make such measurements.

A test plan to characterize radionuclide transport in a VLPC has been published.[147] The plan is comprised of four elements: (1) determination of the chemical and physical forms of key radionuclides (I, Sr, Cs, Te and Ag), (2) sorptivity measurements for VLPC materials of construction, (3) investigation

of coatings with high iodine sorptivities, and (4) integral validation tests. It is suggested that determination of chemical and physical forms could be accomplished in conjunction with post-irradiation heating tests by employing appropriate measurement techniques such as residual gas analyzers and mass spectrometers. Sorptivity measurements would require the use of very low concentrations of radionuclides in the gas phase to be relevant to conditions in the VLPC. Caution would be required in the investigation of paint or surface coating with high iodine sorptivities to avoid the formation of volatile organic iodide species, which could be counter-productive in terms of iodine holdup in the VLPC. Integral validation tests could be performed with a simulated VLPC structure using an in-pile loop or in a hot cell using reactivated irradiated fuel, although no facility of either type currently exists.

A test specification for single-effects fission product transport tests in an out-of-pile loop has been published.[148] The specification covers the principal components of the loop, the radionuclide source, the metal components, and the test measurements. Measurements of radionuclide deposition under steady-state conditions and re-entrainment during transient blowdown conditions are specified for dry and wet (effects of moisture ingress) environments. Radionuclides of interest are Ag, Cs, I, Te, and Sr.

Accomplishments in tritium transport since May 2009 have been provided[149] and include: a test plan for hydrogen permeation tests,[150] fabrication of an experimental apparatus for measuring hydrogen and tritium permeability through high-nickel alloys, and completion of performance verification of the experimental system, including temperature mapping of the heated alloy samples by thermal imaging. The performance of the system was verified by comparison of the measured hydrogen permeability of Incoloy 800H with values reported by several sources in the literature. In addition, data have been obtained on the hydrogen permeability of Haynes 230 and Inconel 617 over the temperature range 650 to 950°C using gas mixtures of 100, 1,000, and 10,000 ppm H_2 in He, and are the first high-temperature permeability data for these alloys. Characterization of samples by metallographic and surface analysis before and after testing has been initiated to understand the effect of surface conditions, such as oxide formation, on hydrogen permeability. A 2D computational fluid dynamics model of the test section is in development to calculate time dependent flow fields, heat transfer in gas and solid, and diffusion of hydrogen/tritium in fluid and solid regions. Work controls and quality assurance requirements for hydrogen and tritium operation have been implemented.

6. REFERENCES

1. ORNL, *Technical Program Plan for the Advanced Gas Reactor Fuel Development and Qualification Program*, ORNL/TM-2002/262, April 2003.
2. G.Bourchardt, B. Hürttlen, and G. Pott, "Experiment FRJ2-P24 Bestrahlungsbericht", Kernsforschugsanlage Jülich GMBH, Report No. ZBB/IB/19/82.
3. Schenk, W., R. Gontard, and H. Nabielek, *Performance of HTR Fuel Samples under High-Irradiation and Accident Simulation Conditions, with Emphasis on Test Capsules HFR-P4 and SL-P1*, Research Center Julich, October 1994.
4. INEEL, Independent Technology Review Group, *Design Features and Technology Uncertainties for the Next Generation Nuclear Plant*, INEEL/EXT-04-01816, June 30, 2004.
5. INL, *Assessment of Next Generation Nuclear Plant (NGNP) Fuel Acquisition Strategies*, INL/EXT-07-12441, Rev. 1, November 2007.
6. S. Muñoz, *Fuel Product Specification*, DOE-HTGR-100209, May 1994.
7. R. R. Hobbins and R. K. McCardell, *Summary of NP-MHTGR Fuel Failure Evaluation*, EGG-NPR-10967, September 1993
8. General Atomics, *MHTGR TRISO-P Fuel Failure Evaluation Report*, DOE-HTGR-90390, Rev. 0, October 1993.
9. INL, "Preliminary AGR Fuel Specification," EDF-4198, Rev. 1, April 2004.
10. R. Gontard and H. Nabielek, *Performance Evaluation of Modern HTR TRISO Fuels*, HTA-IB-05/90, July 1990
11. D. A. Petti, et al., *Key Differences in the Fabrication, Irradiation, and Safety Testing of U.S. and German TRISO-coated Particle Fuel and Their Implications on Fuel Performance*, INEL/EXT-02-00300, April 2002.
12. ASME, NQA-1, "2000 Quality Assurance Requirements for Nuclear Facility Applications with Addenda," 2000.
13. J. Ketterer, *Capsule HRB-16 Preirradiation Report*, General Atomics 906200, Issue 1, September 1981.
14. General Atomics, *Capsule HRB-21 Preirradiation Report*, DOE-HTGR-88357, Rev. C, April 1991.
15. Babcock & Wilcox, "*NPR-MHTGR Performance Test Fuel, Uranium Oxycarbide Fuel Kernels Composite, Lot Number: B10-K-91381*," April 26, 1991.
16. INL, "AGR-1 Fuel Product Specification and Characterization Guidance," EDF-4380, Rev. 7, February 2006.
17. P. E. MacDonald, et al., *NGNP Preliminary Point Design – Results of the Initial Neutronics and Thermal-Hydraulic Assessments*, INEEL/EXT-03-00870, July 2003.
18. C. A. Baldwin, et al., *Fuel Capsule HRB-21 Postirradiation Examination Data Report*, DOE-HTGR-100229, April 1995.

19. 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.

20. *Fuel Performance and Fission Product Behavior in Gas Cooled Reactors*, IAEA-TECDOC-978, November 1977. – Existing Reference in TPP Rev. 2.

21. HTGR, *Radionuclide Design Criteria*, Section 11.1, "Preliminary Safety Information Document for the Standard MHTGR," Volumes 1–6, HTGR-84-024, Amendment B, August 1992.

22. Hanson, D. L., *Test Plan for Characterizing Tritium Transport in a VHTR*, General Atomics, PC-000550, Rev. 0, December 2007.

23. J. M. Kendall, and R. R. Hobbins, "Moisture Ingress from Direct Cycle Steam Generation – Effect on Fuel Performance and Fission Product Technology Development," TEV-583, July 2009.

24. DOE-HTGR-86024, "Preliminary Safety Information Document for the Standard MHTGR", Original document issued October 1985 plus Amendments 1 through 13 (August 1992).

25. R. R. Hobbins, "Moisture Ingress Study Planning," VHTR R&D FY10 Technical Review Meeting, Denver, Colorado, April 27–29, 2010.

26. GA Technologies, Bechtel National, and Combustion Engineering, "Design Data Needs Modular High-Temperature Gas-Cooled Reactor", DOE-HTGR-86-025, Rev. 2, March 1987.

27. M. P. Kissane, "A Review of Radionuclide Behaviour in the Primary System of a Very-High-Temperature Reactor," Nucl. Eng. Design, 239, 3076, 2009.

28. C. Oh, "Air Ingress Experiment and Validation of CFD Code", VHTR R&D FY10 Technical Review Meeting, Denver, Colorado, April 27-29, 2010.

29. INL, *AGR-1 Fuel Product Specification and Characterization Guidance*, EDF-4380, Rev. 7, February 2006.

30. INL, *Preliminary AGR Fuel Specification*, EDF-4198, Rev. 1, April 2004.

31. Kim, T., et al., *Whole-Core Depletion Studies in Support of Fuel Specification for the Next Generation Nuclear Plant (NGNP) Core*, ANL, July 2004.

32. Sterbentz, J., et al., *Reactor Physics Parametric and Depletion Studies in Support of TRISO Particle Fuel Specification for the Next Generation Nuclear Plant*, INEEL/EXT-04-02331, September 2004.

33. C. M. Barnes, *AGR-2 Fuel Specification*, SPC-923, Rev. 3, INL, January 2009.

34. D. W. Marshall, *AGR DTF Fuel and Capsule Component Material Specifications*, SPC-1214, Rev. 0, April 15, 2010.

35. BWXT Certified Data Package, "Certificate of Conformance for NUCO Kernel," February 2004.

36. BWXT Certified Data Package, "G73 Industrial Fuel Fabrication and Development Lot G73D-20-69302," March 2005.

37. BWXT Certified Data Package, "G73 Industrial Fuel Fabrication and Development Lot G73V-20-69303," June 21, 2006.

38. M. Ebner, *Chemistry Improvement for the Production of LEU UCO Fuel Using Manufacturing Scale Equipment – FY2004 Final Report*, INEEL/EXT-04-02372, September 2004.

39. M. Ebner, *Chemistry Improvement Tests for the Production of LEU UCO Fuel Using Manufacturing Scale Equipment – FY2005 Final Report*, August 2005.

40. M. Ebner, C. M. Barnes, *UCO Kernel Fabrication Development for the Advanced Gas Reactor Program*, INL/EXT-07-12247, March 2007.

41. C. M. Barnes, D W. C. Richardson, and M. A. Ebner, "Fabrication Process and Product Quality Improvements in Advanced Gas Reactor UCO Kernels," Proceedings of the 4th International Topical Meeting on High Temperature Reactor Technology, Washington D.C., Paper HTR2008-58039, September 28-October 1, 2008.

42. S. G. Nagley, C. M. Barnes, D. L. Husser, M. L. Nowlin, and W. C. Richardson, "Fabrication of Uranium Oxycarbide Kernels for HTR Fuel," Proceedings of the 5th International Topical Meeting on High Temperature Reactor Technology, Prague, Czech Republic, Paper HTR2010-227, October 18-20, 2010.

43. R. Lowden, et al., *Effects of Deposition Conditions on the Properties of Pyrolytic Carbon Deposited in a Fluidized Bed*, ORNL/TM-2005/533, September 2005.

44. J. Hunn, *Results from ORNL Characterization of German Fuel From EUO 2358-2365 Composite*, ORNL/CF-04/06, April 2004.

45. J. Hunn, *Results from ORNL Characterization HRB-21 Reference Fuel*, ORNL/CF-04/08, August 2004.

46. C. E. A, Finney, et al., *ORNL FY06 Progress Report on FBCVD Process Modeling for the Advanced Gas Reactor Fuel Development and Qualification Program*, ORNL/CF-06/06, September 2006.

47. R. Lowden and J. McLaughlin, *Fluidized Bed Coating of 500 µm Depleted Uranium Oxide Kernels*, ORNL.CF-04/13, Rev. 0, September 2004.

48. R. Lowden, *Fabrication of Baseline AGR-1 Coated-Particle Fuel*, ORNL Milestone Letter Report AGR-1045, December 2005.

49. J. Hunn and R. Lowden, *Data Compilation for AGR-1 Baseline Coated Particle Composite LEU01-46T*, ORNL/TM-2006/019, Rev. 0, April 2006.

50. R. Lowden, *Fabrication of Baseline and Variant Particle Fuel for AGR-1*, ORNL/CF-06/02, Rev. 0, April 2006.

51. J. Hunn, and R. Lowden, *Data Compilation for AGR-1 Variant 1 Coated Particle Composite LEU01-47T*, ORNL/TM-2006/020, Rev. 0, April 2006.

52. J. Hunn, and R. Lowden, *Data Compilation for AGR-1 Variant 2 Coated Particle Composite LEU01-48T*, ORNL/TM-2006/021, Rev. 0, May 2006.

53. J. Hunn, and R. Lowden, *Data Compilation for AGR-1 Variant 3 Coated Particle Composite LEU01-49T*, ORNL/TM-2006/022, Rev. 0, May 2006.

54. J. Hunn, and R. Lowden, *Data Compilation for AGR-3/4 Driver Fuel Coated Particle Composite LEU03-09T*, ORNL/TM-2007/019, Rev. 0, March 2007.

55. A. K. Kercher, J. D. Hunn, and F. C. Montgomery, "Analysis of candidate design-to-fail particles named LEU04-02DTF," ORNL, December 31, 2009.

56. J. Hunn, and J. Miller, *Data Compilation for AGR-3/4 Designed-to-Fail Fuel Particle Batch LEU04-02DTF*, ORNL/TM-2008/193, October 2008.

57. INL, Advanced *Gas Reactor Coater Scale Up Plan*, PLN-1975, February 2006.

58. INL, *Test Plan for Validation of AGR 6-inch Diameter Coater Design*" EDF-7202, September 2006.

59. D. W. Marshall, "*Six-inch TRISO Fuel Coater Design for AGR-2*, EDF-6666, Rev. 1, INL, September 2006.

60. D. W. Marshall, "and C. M. Barnes, *AGR Fuel Development – Coater and Control System Upgrade*, INL-EXT-07-12458, March 2007.

61. C. M. Barnes and D. W. Marshall, *FY2008 Six-inch Diameter Coater Test Report*, INL/EXT-08-14633, September 2008.

62. C. M. Barnes and D. W. Marshall, *FY2009 Particle Fabrication and Coater Test Report*, INL/EXT-09-16545, August 2009.

63. T. Burchell and P. Pappano, *AGR Fuel Compact Development Plan*, ORNL/TM-2004/18, October 2003.

64. P. J. Pappano, *Baseline and Variant Fuel Compact Fabrication Report for AGR-1*, ORNL/TM-2006/528, June 2006.

65. J. Hunn, *Data Compilation for AGR-1 Baseline Compact Lot LEU01-46T-Z*, ORNL/TM-2006/507, Rev. 0, August 2006.

66. J. Hunn, *Data Compilation for AGR-1 Variant 1 Compact Lot LEU01-47T-Z*, ORNL/TM-2006/508, Rev. 0, August 2006.

67. J. Hunn, *Data Compilation for AGR-1 Variant 2 Compact Lot LEU01-48T-Z*, ORNL/TM-2006/509, Rev. 0, August 2006.

68. J. Hunn, *Data Compilation for AGR-1 Variant 3 Compact Lot LEU01-49T-Z*, ORNL/TM-2006/510, Rev. 0, August 2006.

69. P. J. Pappano and J. Hunn, *Update on Over-coating and Compacting Activities for Coated Particles with 425 µm Kernels*, ORNL/TM-2008/009, January 2008.

70. P. J. Pappano et.al., *Final Report on Over-coating and Compacting Activities for Coated Particles with 425 µm UCO and 500 µm UO2 Kernels*, ORNL/TM-2008/091, June 2008.

71. E. Shaber, *AGR Fuel Development Compacting Scale-Up Plan*, PLN-2908, Rev. 0, April 2009.

72. M. Nilles, *Over-coating Studies for Compacting/Matrix Development*, BWED-10-143, Rev. 00, April 21, 2010.

73. Babcock and Wilcox Nuclear Operations Group, *AGR Compacting Pilot Facility Final Design Report*, BWED-10-144, Rev 00, May 18, 2010.

74. G. E. Jellison, Jr., "Optical Characterization of Tristructural Isotropic Fuel Particle Cross-sections Using Generalized Ellipsometry," *Journal of Nuclear Materials*, Vol. 352, pp. 6–12.

75. Y. Katoh, et al., *Strength Evaluation of PyC for TRISO Particles: Development of Equibiaxial Flexural Test*, ORNL/TM-2008/164, September 2008.

76. T. S. Byun, et.al., *Fracture Stress Data for SiC Layers in TRISO-Coated Fuel Particles*, ORNL/TM-2008/167, September 2008.

77. G. Vasudevamurthy and Y. Katoh, *Preliminary Assessment of Irradiation Performance of ZrC in TRISO Coated Particles*, ORNL, November 25, 2009.

78. INL, "Advanced Gas Reactor-1 Pre-Test Prediction Analyses using the PARFUME Code," EDF-5741, Rev. 1, April 2007.

79. J. Hunn, F. Montgomery, and P. Pappano, *Data Compilation for AGR-2 UCO Baseline Compact Lot LEU07-OP1-Z*, Rev 1, ORNL/TM-2009/305, December 2009.

80. J. Hunn, F. Montgomery, and P. Pappano, *Data Compilation for AGR-2 PBMR UO2 Compact Lot LEU08-OP3-Z*, Rev 0, ORNL/TM-2010/070, March 2009.

81. J. Hunn, F. Montgomery, and P. Pappano, *Data Compilation for AGR-2 UCO Variant Compact Lot LEU09-OP2-Z*, Rev 1, ORNL/TM-2010/017, February 2010.

82. J. Hunn, F. Montgomery, and P. Pappano, *Data Compilation for AGR-2 B&W UO2 Compact Lot LEU11-OP2-Z*, Rev 1, ORNL/TM-2010/055, March 2010.

83. J. Hunn, T. Byun, and J. Miller, *Fabrication and Characterization of Sixteen SiC Variants Deposited on the Same IPyC Substrate for Fracture Strength Testing*, ORNL/TM-2009/324, December 2009.

84. J. Hunn, *ORNL Data Compilation for the IAEA CRP-6 Coated Particle Fuel Quality Control Benchmark Exercise*, ORNL/TM-2008/226, November 2008.

85. M. Ebner, *Composition Distribution of Fuel Kernels in LEU Composite 69302*, EDF-5723, Rev. 0, April 13, 2005.

86. C. Barnes, and M. A. Ebner, *Disposition of BWXT Kernel Lot G73D-20-69302*, EDF-5796, Rev 0, April 13, 2005.

87. C. Barnes, *Assessment of BWXT Particle Characterization Capabilities*, EDF-7316, September 19, 2006.

88. Babcock & Wilcox, Industrial Fuel Fabrication and Development Lot G73I-14-69307 (AGR-2 UCO kernel data package), July 30, 2008.

89. Babcock & Wilcox, Industrial Fuel Fabrication and Development Lot G73AA-10-69308 (AGR-2 UO2 kernel data package), December 22, 2008.

90. Babcock & Wilcox, Industrial Fuel Fabrication and Development Lot G73J-14-93072A (AGR-2 UCO coated particles data package), December 9, 2008.

91. Babcock & Wilcox, Industrial Fuel Fabrication and Development Lot G73J-14-93071A, G73J-14-93073A, G73J-14-93074A (AGR-2 UCO coated particles data package), September 12, 2008.

92. Babcock & Wilcox, Industrial Fuel Fabrication and Development Lot G73H-10-93085B (AGR-2 UO2 coated particles data package), March 13, 2009.

93. J. Hunn, *Data Compilation for AGR-2 Baseline Coated Particle Batch G73J-14-93071A*, ORNL/TM-2008/132, October 2008.

94. J. Hunn, *Data Compilation for AGR-2 Baseline Coated Particle Batch G73J-14-93072A*, ORNL/TM-2008/133, October 2008.

95. J. Hunn, *Data Compilation for AGR-2 Baseline Coated Particle Batch G73J-14-93073A*, ORNL/TM-2008/134, October 2008.

96. J. Hunn, *Data Compilation for AGR-2 Baseline Coated Particle Batch G73J-14-93074A*, ORNL/TM-2008/135, October 2008.

97. J. Hunn, *Data Compilation for AGR-2 B&W UO2 Coated Particle Batch G73H-10-93085B*, ORNL/TM-2009/255 (draft), November 2009.

98. J. Hunn, *Data Compilation for AGR-2 PBMR UO2 Coated Particle Batch CPU-L-H101 (LEU08)*, ORNL/TM-2009/096, March 2009.

99. E. Shaber, *Compacting Scale-Up Mass Balance and Process Flow Evaluation*, TEV-596, Rev. 0, August 2009.

100. J. Hunn, F. Montgomery, and P. Pappano, *Data Compilation for AGR-2 UCO Variant Compact Lot LEU06-OP1-Z*, Rev 1, ORNL/TM-2009/304, December 2009.

101. J. Hunn, F. Montgomery, and P. Pappano, *Data Compilation for AGR-2 UCO Baseline Compact Lot LEU07-OP1-Z*, Rev 1, ORNL/TM-2009/305, December 2009.

102. J. Hunn, F. Montgomery, and P. Pappano, *Data Compilation for AGR-2 PBMR UO2 Compact Lot LEU08-OP3-Z*, Rev 0, ORNL/TM-2010/070, March 2009.

103. J. Hunn, F. Montgomery, and P. Pappano, *Data Compilation for AGR-2 UCO Variant Compact Lot LEU09-OP2-Z*, Rev 1, ORNL/TM-2010/017, February 2010.

104. J. Hunn, F. Montgomery, and P. Pappano, *Data Compilation for AGR-2 B&W UO2 Compact Lot LEU11-OP2-Z*, Rev 1, ORNL/TM-2010/055, March 2010.

105. C. M. Barnes, and D. W. Marshall, *Advanced Gas Reactor Coater Scale Up Plan*, PLN-1975, January 31, 2006, Appendix 1.

106. J. T. Maki, *Preliminary AGR-1 Irradiation Test Specification*, EDF-4187, Rev. 0, INL, September 2003.

107. J. T. Maki, *AGR-1 Irradiation Test Specification*, EDF 4731, Rev. 1, INL, July 2004.

108. INL, *Requirements for Design of the Advanced Gas Reactor Experiment AGR-1 for Irradiation in the Advanced Test Reactor*, TFR-249, Rev. 1, December 2004.

109. INL, *Temperature Control and Off Gas Monitoring Systems for Advanced Gas Reactor Experiment AGR-1*, TFR-248, March 2005.

110. INL, *Final Design Review Closure for the Advanced Gas Reactor (AGR) AGR-1 Capsule and Fission Product Monitoring System*, CCN 202337, August 2005.

111. G. S. Chang, *Final Neutronics Evaluations of the AGR-1 Experiment in ATR East Position B-10*, EDF-5683, INL, September 22, 2005.

112. D. A. Petti and J. T. Maki, *Irradiation Temperature for AGR-1*, EDF-5016, INL, June 2004.

113. A. J. Palmer, "*Thermocouple Recommendations for the AGR-1 Test*, EDF-6809, INL, May 2006.

114. INL, "AGR Ready to Insert," Quality Assurance Surveillance Report No. 2307, September 2006.

115. G. D. Roberts (Gary.Roberts@inl.gov), *AGR-1 ORNL Data Package Review Complete*, Email to D. Petti dated September 25, 2006.

116. R. G. Ambrosek, *Experiment Safety Analysis for Advanced Gas Reactor Capsule Experiment (AGR-1) Facility*, INL, September 2006.

117. J. M. Kendall, *The AGR-1 Irradiation – Objectives, Success Criteria and Risk Management*, INL/EXT-06-11487, Rev. 0, June 2006.

118. J. T. Maki, *AGR-1 Irradiation Experiment Test Plan*, INL/EXT-05-00593, Rev. 1, September 2006.

119. INL, *Advanced Gas Reactor–1 Pre-Test Prediction Analyses using the PARFUME Code*, EDF-5741, Rev. 1, April 2007.

120. J. T. Maki, *AGR-1 As-Run Analysis Status for FY-07*, INL/EXT-07-13630, December 2007.

121. M. A. Pope, *AGR-1 Irradiation Test Final As-Run Report*, INL/EXT-10-18097, Rev. 1, April 2010.

122. J. T. Maki, *AGR-2 Irradiation Test Specification*, SPC-1064, Rev. 0, May 2009.

123. P. Demkowicz, *Assessment of INL Capabilities for AGR Capsule Postirradiation Examination*, INL/EXT-06-11558, July 2006.

124. R. N. Morris, *An Assessment of ORNL PIE Capabilities for the AGR Program Capsule Postirradiation Examination*, ORNL/TM-2006-526, June 2006.

125. P. Demkowicz, *AGR-1 Postirradiation Examination Test Requirements*, INL/EXT-06-12051, December 2006.

126. P. Demkowicz, *NGNP/AGR Fuel PIE Facility Development Plan*, INL/EXT-06-12060, December 2006.

127. P. Demkowicz, *AGR-1 Post irradiation Examination Plan*, PLN-2828, Rev. 1, July 2010.

128. P. Demkowicz, *AGR-1 Post irradiation Examination Plan*, PLN-2828, Rev. 0, March 2010.

129. G. K. Miller, D. A. Petti, J. T. Maki, and D. L. Knudson, "PARFUME Theory and Model Basis Report," INL/EXT-08-14497, September 2009.

130. D. A. Petti, et al., *Development of Improved Models and Designs for Coated-Particle Gas Reactor Fuels*, INEEL/EXT-05-02615, December 2004.

131. "Redlich-Kwong equation of state," http://www.absoluteastronomy.com/encyclopedia/E/Eq/Equation_of_state.htm.

132. D. L. Knudson, *Status of PARFUME Accident Condition Modeling*, EDF-8375, September 2007.

133. G. K. Miller, and D. L. Knudson, *PARFUME Benchmark for Coated Particle Fuel Performance Under Normal Operation*, EDF-6036, Rev. 0, June 2005.

134. D. L. Knudson and G. K. Miller, "Accident Condition Benchmark Calculations using the PARFUME Computer Code – A Final Report," Idaho National Laboratory, ECAR No. 728, August 27, 2009.

135. G. K. Miller and W. F. Skerjanc, "Status Report on Incorporating Zirconium Carbide Modeling Capabilities within the PARFUME Code," Idaho National Laboratory, ECAR No. 740, August 27, 2009.

136. W. F. Skerjanc, "Fuel Particle Failure Probability Response Functions for 425 µm UCO and 500 µm UO2 Fuel Particles," Idaho National Laboratory, ECAR No. 741, September 9, 2009.

137. K. D. Hamman, "AGR-2 Pre-Test Prediction Analyses using the PARFUME Code for U.S. Fuel Particles," ECAR-1020, June 1, 2010.

138. R. R. Hobbins, *Review of Fission Product Transport Research*, EDF 5072, Rev. 4, September 2007.

139. D. L. Hanson, *HFR-B1 Final Summary Report*, General Atomics, PC-000529, Rev. 0, April 2006.

140. D. L. Hanson, *The HFR-B1 Capsule 1 Data Base and its Suitability for Use in Code Validation*, General Atomics, PC-000519, Rev. 0, May 2004.

141. D. L. Hanson, *Recommendations for Fission Product Transport Irradiation Tests*, General Atomics, PC-000527, Rev. 0, September 2005.

142. D. L. Hanson, *Test Specification for Fission Product Sorption Measurements*, General Atomics, 911095, Rev. 0, September 2006.

143. M. Presley and D. Hanson, *Analysis of COMEDIE SR1 Plateout Data with the PADLOC Code*, General Atomics, 911061, Rev. 0, August 2004.

144. D. L. Hanson, *Test Plan for Single-Effects, Fission Product Plateout and Liftoff Tests*, General Atomics, PC-000526, Rev. 0, April 2005.

145. D. L. Hanson, *Requirements for an In-Pile Fission Product Transport Loop*, General Atomics, PC-000522, Rev. 0, December 2004.

146. Hanson, D. L., and J. M. Bolin, *Radionuclide Transport in a Vented Low-Pressure Containment*, General Atomics, PC-000541, Rev. 0, April 2007.

147. D. L. Hanson, "Test Plan to Characterize Radionuclide Transport in a Vented Low-Pressure Containment," General Atomics, PC-000573, Rev. 0, September 2008.

148. D. L. Hanson, "Test Specification for Fission Product Transport Tests in an Out-of-Pile Loop," General Atomics, Report No. 911159, Rev. 0, January 2009.

149. P. Calderoni, "Hydrogen Permeation Studies," VHTR R&D Technical Review Meeting, April 27-29, 2010, Denver, Colorado.

150. P. Calderoni, "High-Temperature Hydrogen Permeation through Nickel Alloys, PLN-3479, Idaho National Laboratory, April 2010.