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ORNL/TM-2002/262

Technical Program Plan for the Advanced Gas Reactor Fuel Development and Qualification Program

Advanced Gas Reactor Team

April 2003

UT-BATTELLE

ORNL-27 (2-03)

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ORNL/TM-2002/262

TECHNICAL PROGRAM PLAN FOR THE ADVANCED GAS REACTOR FUEL DEVELOPMENT AND QUALIFICATION PROGRAM

DOE Program Manager—M. A. Feltus
INEEL Lead—F. H. Southworth
ORNL Lead—G. L. Bell
Plan Integrators—G. L. Bell, R. R. Hobbins,[1] J. M. Kendall,[7] and F. J. Homan[2]

Fuel Manufacture

J. J. Saurwein[3] (Lead) E. L. Shaber[5]
J. A. Halfinger[4] R. A. Lowden[6]

Fuel and Materials Irradiation

J. M. Kendall[7] (Lead) D. A. Petti[5]
D. W. McEachern[3] B. S. Cowell[6]

Safety Testing and PIE Data Needs

R. N. Morris[6] (Lead) R. R. Hobbins[1]
W. J. Scheffel[3]

Fuel Performance Modeling

D. A. Petti[5] (Lead) M. B. Richards[3]

Fission Product Transport and Source Term

R. R. Hobbins[1] (Lead) D. Hanson[3]
R. N. Morris[6]

[1]RRH Consulting, consultant to INEEL
[2]Consultant to ORNL
[3]General Atomics
[4]BWX Technologies
[5]Idaho National Environmental and Engineering Laboratory
[6]Oak Ridge National Laboratory
[7]Global Virtual, LLC

April 2003

Prepared by the
OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37831
managed by
UT-BATTELLE, LLC
for the
U.S. DEPARTMENT OF ENERGY
under contract DE-AC05-00OR22725

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LIST OF ACRONYMS

ADUN	acid-deficient uranyl nitrate
AFCI	Advanced Fuel Cycle Initiative
AGR	advanced gas reactor
Am	americium
ANL-W	Argonne National Laboratory–West
ASME	American Society of Mechanical Engineers
ATR	Advanced Test Reactor (INEEL)
AVR	Arbeitsgemeinschaft Versuchsreaktor
B&W	Babcock & Wilcox
BISO	buffer and isotropic pyrocarbon
BWXT	BWX Technologies (formerly B&W)
CCCD	core conduction cooldown
CCCTF	Core Conduction Cooldown Test Facility
COMEDIE	In-pile loop facility in the SILOE Test Reactor (Grenoble, France)
CPL	In-pile loop facility in the PEGASE Test Reactor (Cadarache, France)
Cs	cesium
CTE	coefficient of thermal expansion
CVD	chemical vapor deposition
DF	decontamination factor
DOE	U.S. Department of Energy
DTF	designed to fail
DUO_2	depleted (<0.71% ^{235}U) uranium dioxide
EAB	exclusion area boundary
ES&H	environmental, safety, and health
EU	European Union
FIMA	fissions per initial metal atom
FSV	Fort St. Vrain
FY	fiscal year
GA	General Atomics
GCR	gas-cooled reactor
Gen-IV	Generation IV
GT-MHR	Gas-Turbine Modular Helium Reactor (Brayton-cycle plant)
HEU	highly enriched uranium (usually ~93%)
HFIR	High Flux Isotope Reactor (ORNL)
HFR	High Flux Reactor (Petten)
HMTA	hexamethylene tetramine
HRB	HFIR removable beryllium
HTGR	high-temperature gas-cooled reactor (a steam-cycle plant)
HTR	High-Temperature Reactor
HTTR	High-Temperature Test Reactor
INEEL	Idaho National Environmental and Engineering Laboratory
INERI	International Nuclear Energy Research Initiative
IMGA	Irradiated Microsphere Gamma Analysis
IPyC	inner pyrolytic carbon
IR	infrared
JAERI	Japan Atomic Energy Research Institute
kW	kilowatt
LEU	low-enriched uranium
LEU UCO	low-enriched (0.71% < ^{235}U content < 20%) uranium mixture of UO_2 and UC_2 herein referring to 19.9% ^{235}U unless otherwise specified

LWR	light-water reactor
MHTGR	modular high-temperature gas-cooled reactor
MHR	modular helium reactor
MIT	Massachusetts Institute of Technology
MW	megawatt
NDE	nondestructive examination
NE	DOE Office of Nuclear Energy, Science, and Technology
NERI	Nuclear Energy Research Initiative
NQA	National Quality Assurance
NOC	normal operating conditions
NRC	U.S. Nuclear Regulatory Commission
NPR	New Production Reactor (an application for producing tritium)
NTD	Near-Term Deployment
NUCO	natural (0.71% ^{235}U) uranium mixture of UO_2 and UC_2
NUO_2	natural (0.71% ^{235}U) uranium dioxide
OGL	Oarai Gas Loop [Japanese in-pile loop]
O&M	operations and maintenance
OPyC	outer pyrolytic carbon coating in a TRISO particle
ORNL	Oak Ridge National Laboratory
PAG	Protection Action Guide
PBR	pebble-bed reactor
PBMR	pebble-bed modular reactor
PMR	prismatic modular reactor
PC-MHR	plutonium consumption modular helium reactor
PGA	particle gas analyzer
PIE	postirradiation examination
PyC	pyrolytic carbon
QA	quality assurance
QAP	quality assurance plan
QC	quality control
R/B	release-to-birth ratio
RES	Office of Nuclear Regulatory Research
R.F.	Russian Federation
R&D	research and development
SAXS	small-angle X-ray scattering
SEM	scanning electron microscope
SiC	silicon carbide
SR	shear ratio
TBD	to be determined
TCE	trichloroethylene
TRIGA	Test, Research, Isotopes, General Atomics (a test reactor)
TRISO	A multilayer coating, on a spherical kernel, made up of a low-density pyrocarbon (buffer) layer surrounded by an IPyC, surrounded by a SiC, surrounded by an OPyC
UCO	mixture of UO_2 and UC_2
UV	ultraviolet
VHTR	Very High Temperature Reactor
VLPC	vented low-pressure containment
V&V	verification and validation
WBS	Work Breakdown Structure
WDX	wavelength dispersive X-ray spectroscopy

EXECUTIVE SUMMARY

The advanced gas reactor (AGR) options addressed in this plan are based on what has been generally referred to as high-temperature gas-cooled reactor (HTGR) technology. The primary distinguishing features of HTGRs are the use of helium coolant, a low-power-density ceramic core capable of withstanding very high temperatures, and coated-particle fuel. The HTGR-technology-based options are an important element of the Department of Energy's (DOE's) program to revitalize the nuclear power generation option in the United States in support of the National Energy Policy. The AGR power plant and process heat concepts offer the prospect of energy production at a competitive cost in smaller unit sizes. If successfully developed and deployed, AGRs will serve a central role in meeting future energy needs with minimal environmental emissions in the United States and internationally. Moreover, the DOE Generation IV program has identified the very-high-temperature gas-cooled reactor (VHTR) concept as uniquely suited for producing hydrogen without the consumption of fossil fuels or the emission of greenhouse gases. DOE has selected the VHTR system for the Freedom Power Project, a project to demonstrate emissions-free nuclear-assisted hydrogen production by 2015.

Two HTGR power reactors, the 40-MW(e) Peach Bottom Unit 1 and the 330-MW(e) Fort St. Vrain, were built and operated in the United States in the 1960s and 1970s. In the 1980s, increasing recognition of the organizational and operational complexities arising from reliance on powered, active safety systems fostered the development of HTGR-based concepts that emphasized inherent passive features to meet public safety requirements. There are currently two basic design categories for AGRs: the pebble-bed reactor (PBR) and the prismatic reactor (PMR). The PBR uses fuel particles agglomerated into billiard-ball-size graphite spheres that slowly move through the reactor core in a continuous refueling process. The PMR uses fuel particles agglomerated into cylindrical rods that are loaded into hexagonal-shaped graphite blocks that make up the reactor core, which is periodically refueled in a batch process. Modern design concepts in both of these categories place a greater reliance on retention of fission products within the coated-particle fuel, which results in higher standards for fuel performance and reliability than were required of fuel for earlier HTGR plants.

This AGR Fuel Development and Qualification Program Plan was developed to address the following overall goals:

- Provide a baseline fuel qualification data set in support of the licensing and operation of the Generation IV Very High Temperature Gas-Cooled Reactor (VHTR). Gas-reactor fuel performance demonstration and qualification comprise the longest duration research and development (R&D) task for VHTR feasibility. The baseline fuel form is to be demonstrated and qualified for a peak fuel centerline temperature of 1250°C.
- Support near-term deployment of an AGR for commercial energy production in the United States (2015) by reducing market entry risks posed by technical uncertainties associated with fuel production and qualification.
- Utilize international collaboration mechanisms to extend the value of DOE resources.

These goals are to be addressed by implementing a coated-particle fuel development and qualification program that supports both the PMR and PBR fuel designs in the early stages. As the program proceeds, the effort dedicated to each design will be proportionate with its associated level of industry interest and U.S. commitment. As discussed in Appendix F, there is a broad body of international data applicable to the PBR concepts; additional development is under way in China, South Africa, and the European Union, with results available to the United States via international collaboration. While these data are relevant for the PMR designs, they are not sufficient to address specific prismatic-core manufacturing parameters and the typically higher PMR fuel performance requirements (i.e., higher burnup, operating temperature and temperature gradients, and fast neutron exposure). Thus, early effort will focus on manufacturing and testing the fuel design used in the PMR concepts, using the most recent

fuel product specification* as a starting point. Irradiation, safety testing, and postirradiation examination (PIE) plans are based on supporting fuel development and qualification in an integrated manner. If an industry initiative based on a pebble-bed design materializes in the future, the irradiation and postirradiation activities would be expanded or redirected, as necessary, starting with a preliminary plan developed for PBR fuel qualification.†

Preliminary operating conditions and performance requirements for the fuel will be generated based on previously completed PMR work and applicable information on the VHTR.‡ Information from future VHTR development activities will be used for the review and modification of fuel product specifications and irradiation test specifications as it becomes available.

Facilities and personnel experienced in the activities necessary to address the program goals remain in place within the United States, primarily at the Oak Ridge National Laboratory (ORNL) and the Idaho National Engineering and Environmental Laboratory (INEEL). This experience and knowledge of facility status and capabilities have been incorporated by the direct involvement of personnel from ORNL and INEEL in the development of this plan. In addition, General Atomics (GA) has provided input regarding PMR fuel performance requirements and perspectives from its prior experience in fuel development, fuel fabrication, and in fuel-related analytical capabilities needed to support licensing interactions. BWX Technologies (BWXT) has also contributed based on its experience and capabilities for fuel kernel production and particle coating.

Many of the individuals responsible for the development of this plan were directly involved in the production and testing of previous U.S. fuel for the Modular High Temperature Gas-Cooled Reactor (MHTGR) and the New Production Reactor (NPR), 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 U.S. coated-particle fuel experience and is generally consistent with the recommendations arising from that experience.

The plan has been developed based on the following major program elements:

- **Fuel manufacture**—This element addresses the work necessary to produce coated-particle fuel that meets fuel performance specifications and includes process development for kernels, coatings, and compacting; quality control (QC) methods development; scale-up analyses; and process documentation needed for technology transfer. This effort will produce fuel and material samples for characterization, irradiation, and accident 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; that work will be conducted during the later stages of the program in conjunction with cosponsoring industrial partners.
- **Fuel and materials irradiation**—The fuel and materials irradiation activities will provide data on fuel performance under irradiation as necessary to support fuel process development, to qualify fuel for normal operation conditions, and to 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 capsules have been defined to provide the necessary data and sample materials.
- **Safety testing and PIE**—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 manufacture effort by providing feedback on the performance of kernels, coatings, and compacts. Data from PIE and accident testing will supplement the in-reactor measurements [primarily fission gas release-to-birth ratio (R/B)] as necessary to demonstrate

*DOE-HTGR-100209, "Fuel Product Specification," May 1994.

†PBMR Fuel Qualification Test Program, March 2002, NRC Adams Accession No. ML020800192.

‡*A Technology Roadmap for Generation IV Nuclear Energy Systems*, U.S. DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum, December 2002. The primary functions of the VHTR are production of hydrogen and process heat.

compliance with fuel performance requirements and support the development and validation of computer codes.

- **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 does not address the release of fission products from the fuel particle, although it considers the effect of fission product chemical interactions with the coatings, which can lead to degradation of the coated-particle properties. Computer codes and models will be further developed and validated as necessary to support fuel fabrication process development and plant design and licensing.
- **Fission product transport and source term**—This element will address the transport of fission products produced within 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 necessary to support plant design and licensing.

Detailed task breakdowns have been developed for each of these program elements, along with cost and duration estimates, deliverables, and interfaces with the other program elements. A detailed discussion of the program elements and their corresponding tasks is provided in the report and further expanded in the appendices. These inputs have been used to develop an integrated program schedule and cost projection based on adequate funding starting in October 1, 2002 (first day of FY 2003). The integrated schedule, taken to the first level below the program element level, is shown in Fig. ES.1. The cost estimates at the program element level are shown in Fig. ES.2. Early program activities are 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. Low 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.

As shown in Fig. ES.3, only a small fraction of the estimated program resources is needed for fuel process development (including the supporting irradiation, safety testing, and PIE), and a smaller fraction is needed for installation of new facilities and equipment. The majority of the resources are devoted to producing data and developing and validating analysis methods in support of AGR licensing, with additional effort directed toward development of fuel mass production technology. This focusing of resources on activities directly supporting AGR deployment is possible because the plan builds heavily on the substantial body of HTGR fuel technology and existing facilities that have been developed in the United States during the past several decades, augmented by relevant international experience and data.

An underlying theme for the work scope presented in this plan is the need to develop a fundamental understanding of the relationship between the fuel fabrication process and fuel product properties and irradiation performance. Fuel performance modeling and analysis of fission product behavior in the primary circuit are also included in this program plan. 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 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.

In the development of this plan, priority has been given to the early activities in support of near-term execution of the plan. Issues associated with longer term activities have been identified but will need to be addressed during the early execution phase and factored into the overall planning. These issues are not expected to affect the basic logic of the plan but will affect the details of its execution. It is expected that the plan will be periodically updated to reflect additional knowledge and the results of ongoing and completed work. The execution of this technical plan will have to 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. The program plan anticipates industrial participation for full execution.

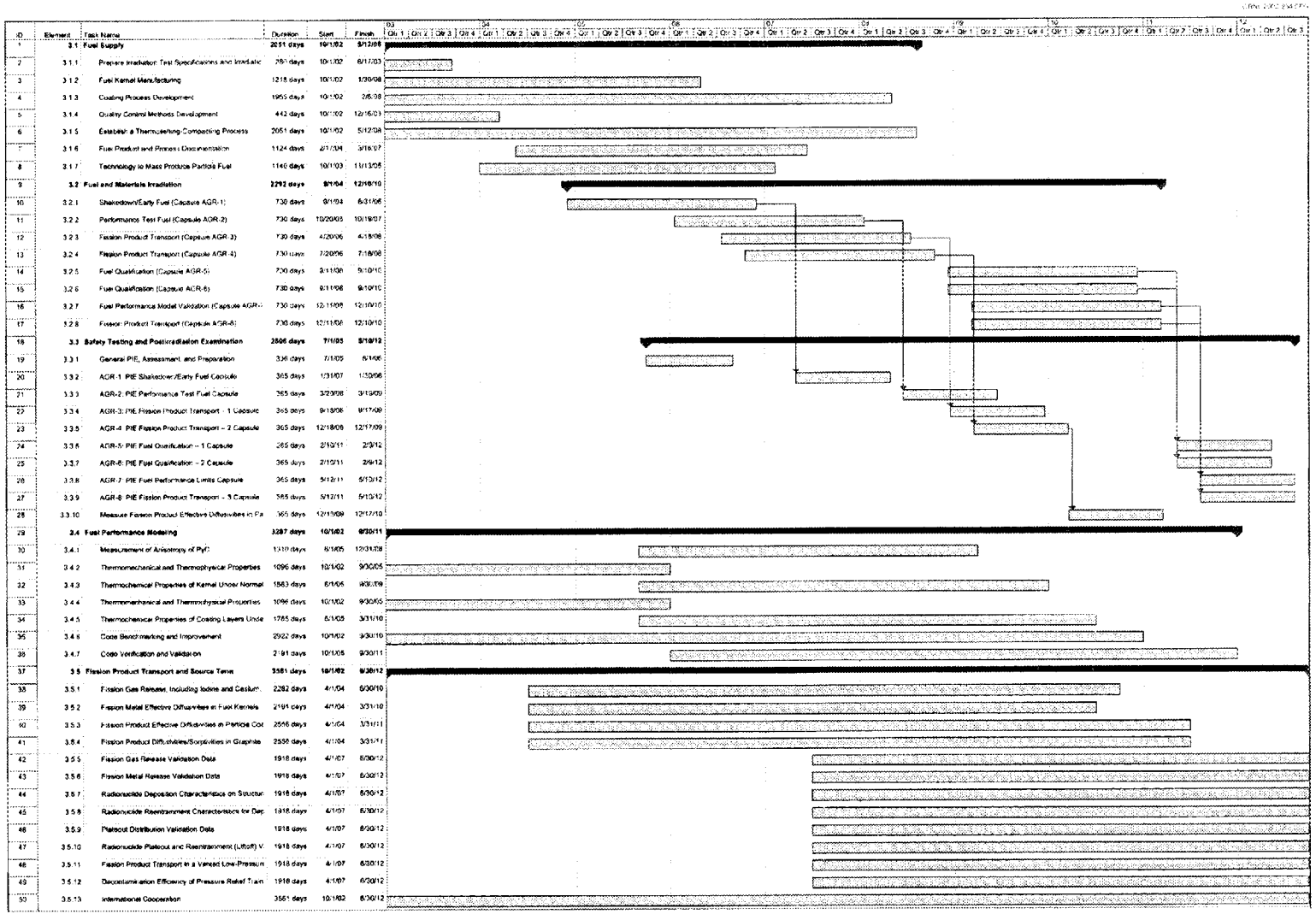

Fig. ES.1. Integrated program schedule.

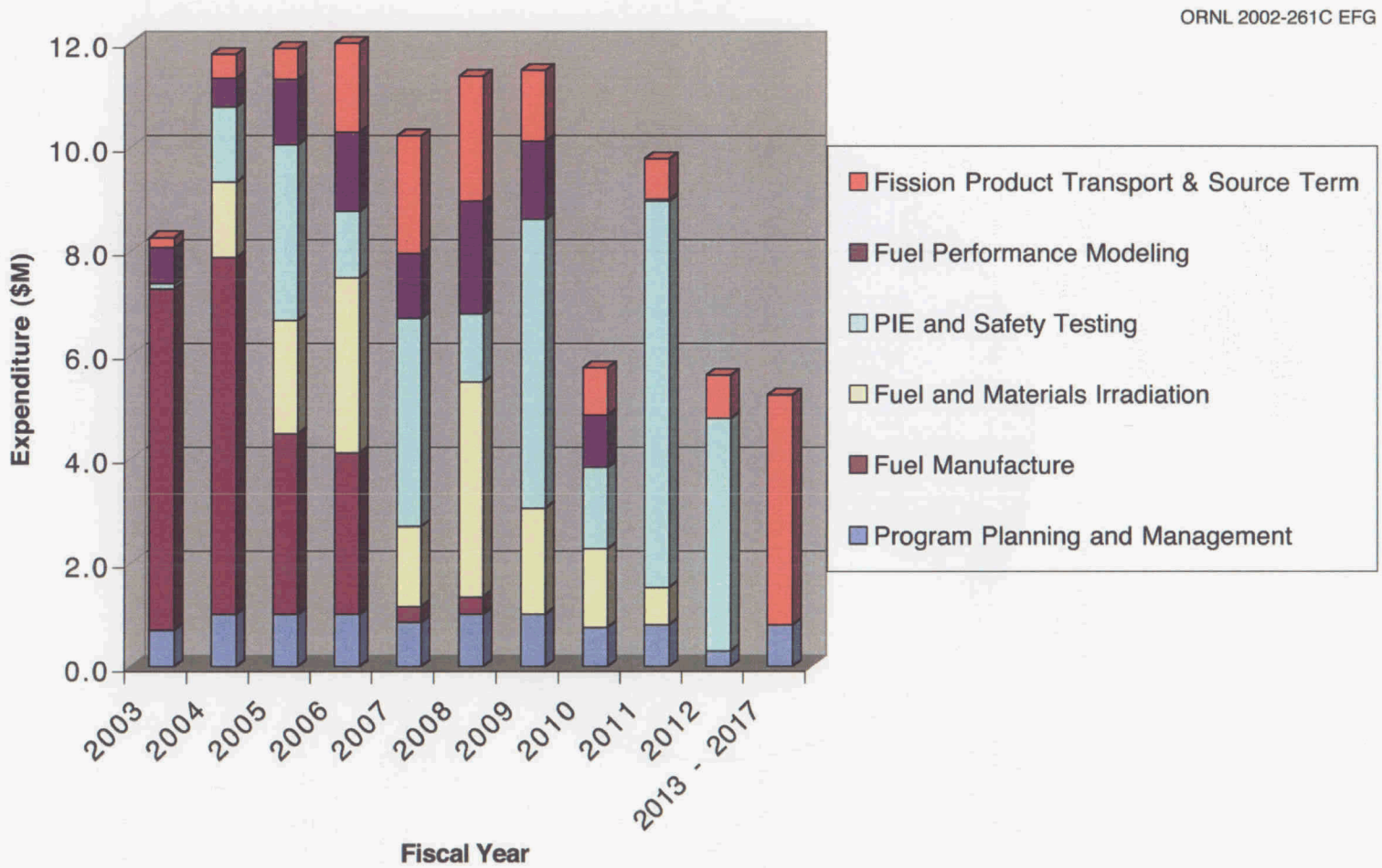

Fig. ES.2. Program expenditure projections.

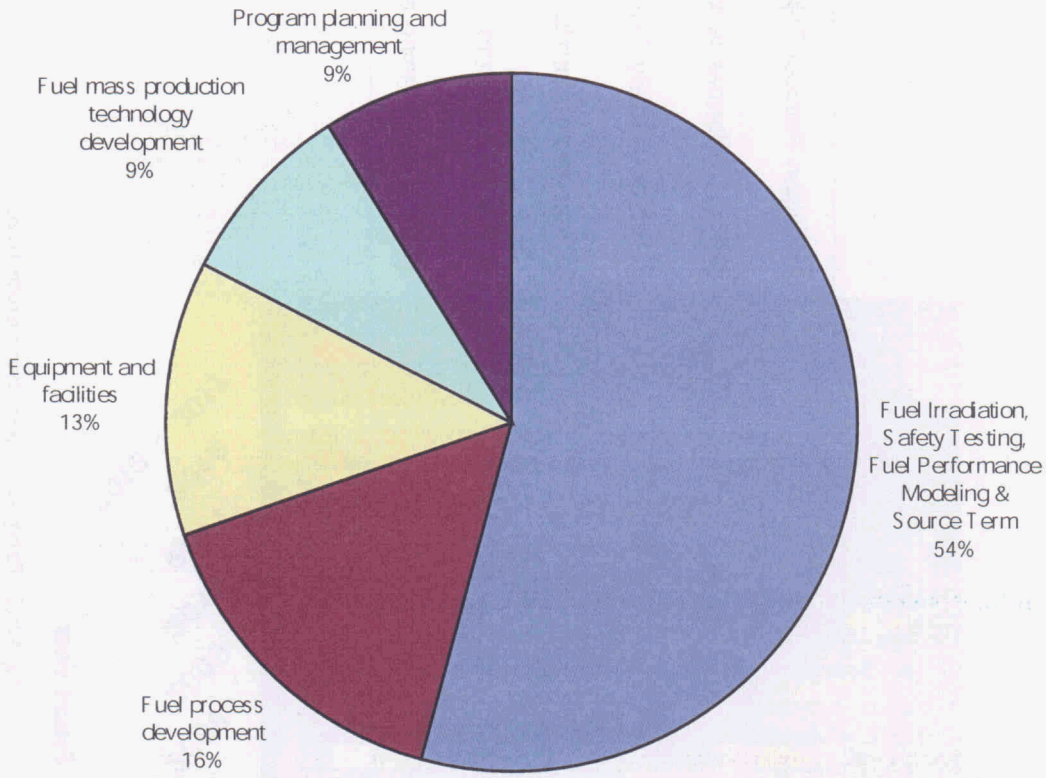

Fig. ES.3. Breakdown of program costs.

1. INTRODUCTION

The Department of Energy (DOE) has initiated programs to revitalize nuclear power generation growth in the United States, in support of the National Energy Policy. Principal among these is the "Generation IV Nuclear Energy Systems Roadmap"[1] (Gen-IV), which defines a technology development path for deployment of nuclear reactor systems by 2030. The goals for the Gen-IV systems are intended to stimulate development, which will result in nuclear reactor systems that excel in sustainability, safety, and economic competitiveness. Among the Gen-IV concepts, the Very High Temperature Reactor System (VHTR) is the nearest-term hydrogen production system, estimated to be deployable by 2020.

A complementary program, Near-Term Deployment (NTD), identified reactor system concepts that might be deployed within the next 10 years and could serve as a bridge to the most promising Gen-IV concepts. Two gas-cooled reactors (GCRs), a pebble-bed reactor (PBR), and a prismatic modular reactor (PMR), were identified by the NTD program as being possibly deployable within the next 10 years.

In the Gen-IV and NTD analyses, the GCR concepts excelled in meeting Gen-IV safety goals. The designs were oriented to provide natural safety, which prevents core damage under all design basis accidents and presently envisioned severe accidents. The principle that guided their design concepts was to naturally maintain core temperatures below fission product release thresholds under all accident scenarios. This level of fission product retention, or fuel performance, 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 level, however, is predicated on exceptionally high fuel fabrication quality and predictability. The Germans produced and demonstrated high-quality fuel for their PBRs in the 1980s. No U.S. manufactured fuel has exhibited equivalent performance. In all Gen-IV and NTD GCR concepts, fuel development and qualification were identified as essential early development needs to assure concept viability.

In fiscal year (FY) 2002, the Office of Nuclear Energy, Science, and Technology (NE) initiated development of a plan for an advanced gas reactor (AGR) fuel development and qualification program for coated-particle fuel. This technical program plan for an "Advanced Gas Reactor Fuel Development and Qualification Program" defines technical activities to support the licensing and operation of a VHTR in the United States.

The design specifications for the VHTR are not yet well-defined, but the maximum burnup envisioned in the VHTR is in the range 150–200 GWd/MTHM, or 16.4–21.8% fissions per initial metal atom (FIMA).[1] The composition of the fuel kernel and the fuel element geometry are open items. Although the Germans have demonstrated excellent performance of SiC TRISO-coated UO_2 particle fuel up to about 10% FIMA and 1200°C, UO_2 fuel is known to have limitations with respect to CO formation and kernel migration at the high burnups, power densities, temperatures, and temperature gradients that may be needed in the VHTR design. With 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 (in irradiation test FRJ-P24) gives added confidence that high-quality SiC TRISO-coated UCO fuel can be made and its performance statistically demonstrated. Thus, SiC TRISO-coated UCO has been chosen as the baseline fuel to be fabricated and tested in this program in support of the VHTR. A PMR fuel element design has been chosen to be consistent with past fuel fabrication experience in the United States and to be complementary to the ongoing PBR research and development (R&D) elsewhere in the world. If a PBR design should be developed for the VHTR, the UCO fuel could be compacted into spherical elements to fit this design. Similarly, if ongoing R&D in China, Europe, or South Africa should indicate that UO_2 fuel could operate successfully at sufficiently extended burnups, the program could be modified to test this fuel to VHTR conditions, as well.

In addition to excellent fission product retention during normal operation at high burnups and high temperatures, the VHTR 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 enhanced cesium releases at burnups ≥14% FIMA,[2] so accident performance testing is an important element of the program.

The TRISO-coated UCO fuel specification,[3] utilizing SiC as the primary fission product containment layer and developed in response to the extensive evaluations[4,5] of the fuel failures in the New Production Reactor (NPR) and HRB-21 irradiations, will be the starting point for the fuel development work. It is expected that this fuel will permit high-burnup operation (22% FIMA) at fuel temperatures up to 1250°C with fast fluences to 4×10^{25} neutrons/m^2, supporting approach to the lower range of VHTR core outlet temperatures (850°C–1000°C); but new coatings, such as ZrC, may be required for higher core outlet temperatures. Work on ZrC as a fuel particle coating is planned as part of the DOE's Advanced Fuel Cycle Initiative (AFCI) program.

This program plan identifies the R&D needed in the areas of fuel manufacture, fuel and materials irradiation, safety testing and postirradiation examination (PIE), fuel performance modeling, and fission product transport and source term. The plan includes an integrated schedule and budget for the work required to develop and transfer a coated-particle fuel fabrication capability in the United States. In the late 1980s the Germans 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 is a fuel design[3] based on the U.S. modular high-temperature gas-cooled reactor (MHTGR) and NPR programs, combined with the successful German coating process. The basic structure of this program is to

1. acquire German coating process information and German fuel and material property data;
2. perform additional characterization of the coating properties of the German particles as needed;
3. 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 VHTR particle design;
4. reestablish basic quality control (QC) capability for coated-particle fuel and develop new QC methods (as required) for enhanced characterization of kernels and coatings;
5. develop an improved fuel compact fabrication process using a thermosetting matrix;
6. produce 2-in.-long by 0.5-in.-diam compacts for shakedown irradiation testing;
7. **assemble a protoypic size coater and set/test the reference coating process parameters with the selected equipment;**
8. **produce fuel with a prototypic size coater and prototypical compacting equipment for fuel performance testing;**
9. perform irradiation testing, accident simulation, safety testing, and PIE of the reference fuel and selected variants to obtain irradiation and accident condition performance data;
10. **produce fuel with a prototypic-size coater and prototypical compacting equipment for fuel qualification testing;**
11. perform irradiation testing, accident condition testing, and PIE of the qualification test fuel to demonstrate that the reference fuel meets VHTR fuel performance requirements for normal operating conditions and accident conditions;
12. reduce fuel manufacturing risk by developing automated QC and fuel process feedback technologies that can be implemented in a fuel manufacturing pilot plant to support low-cost fuel production; and
13. transfer the fuel fabrication and QC technology to industrial organizations interested in deploying an AGR.

An underlying theme in the work scope is the need to improve the understanding of the relationship between the fuel fabrication process, fuel product properties, and irradiation performance. Fuel performance modeling and analysis of fission product behavior in the primary circuit 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 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.

As noted in the schedules presented in this plan, items 1–12 can be accomplished within the first 10 years of the program. Item 13, the modeling, and fission product work will extend beyond the first 10 years. Industrial participation and cost-sharing will be required in later years for full execution of the plan. The fuel performance modeling and fission product behavior testing and modeling activities, while closely linked to the fuel manufacture and testing tasks, are part of the Nuclear Regulatory Commission (NRC) acceptance efforts for this program and will be cost-shared with industry. However, actual 10 CFR Part 50, Appendix B, licensing activities to meet NRC licensing requirements will be performed by the fuel vendors at their own expense. The modeling and fission product behavior tasks will be started during this program, but they will be completed well after the fuel manufacture and testing goals have been met. Because plant licensing may also be a cost-shared activity between DOE and industry, it is anticipated that completion of these activities will be part of the cost-sharing arrangements to be negotiated with industry. The cost-shared elements of the program will provide fundamental data for NRC Office of Research acceptance and preliminary pre-application information for NRC's use. The actual licensing activities conducted by the NRC Office of Nuclear Reactor Regulation will be undertaken by an industrial fuel and/or reactor vendor for their specific fuel fabrication processes and fuel performance verification processes.

A logical extension of the irradiation and safety testing activity is to establish the operating margins for the fuel. For VHTR fuel, this would mean measuring the fuel performance at temperature, fast neutron exposure, and burnup levels at which the fuel begins to fail in significant quantity either during normal operation or under accident conditions. The irradiation capsule designated AGR-7 will be designed so that some measurable level of fuel failure is expected to occur in support of both fuel performance model validation and margin testing. Because margin testing is generally a licensing issue, it is anticipated that the licensee or other interested organizations would fund additional margin testing, if needed. While irradiation capsules devoted exclusively to accident margin testing by NRC are not included in this program plan, the infrastructure required to execute an NRC accident margin testing program will be readily available as a result of planned irradiation tasks.

2. GOALS, ASSUMPTIONS, AND REQUIREMENTS

An overall set of programmatic goals, assumptions, and requirements was developed to guide preparation of this Technical Plan; this set is presented here. In the preparation of this Technical Plan, the scope was subdivided into five program elements:

1. fuel development and manufacture,
2. fuel and materials irradiation,
3. safety testing and PIE,
4. fuel performance modeling, and
5. fission product transport and source term.

More detailed goals, assumptions, and requirements associated with each of the five program elements are documented in Appendices A–E.

2.1 OVERALL PROGRAM GOALS

- Provide a baseline fuel qualification data set in support of the licensing and operation of the Gen-IV VHTR. Gas-reactor fuel performance demonstration and qualification comprise the longest duration R&D task for VHTR feasibility. The baseline fuel form is to be demonstrated and qualified for a peak fuel centerline temperature of 1250°C.
- Support near-term deployment of an AGR for commercial energy production in the United States (2015) by reducing market entry risks posed by technical uncertainties associated with fuel production and qualification.
- Utilize international collaboration mechanisms to extend the value of DOE resources.

Fuel qualification is herein defined as the demonstration of the robust performance and efficacy of the reference coated-particle fuel form through presentation of experimental data and analysis results. This fuel qualification effort is meant to support the Nuclear Regulatory Commission Office of Nuclear Regulatory Research (NRC-RES) in its preapplication review efforts for the VHTR concept and to support the NRC in its eventual issuance of the Gen-IV VHTR license.

2.2 OVERALL PROGRAM ASSUMPTIONS

- Government and industry sponsors recognize that a stable, long-term, disciplined fuel development 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 (i.e., AVR 21-2) meets the needs of the U.S. program.
- Proposed AGR designs will impose more demanding service conditions than the German High-Temperature Reactor (HTR) Module and require testing of a fuel based on the U.S. MHTGR design and the German coating process.
- It is technically feasible to reestablish a production capability equivalent to the German capability in the United States at reasonable cost and on a schedule consistent with initial plant deployment schedule(s).
- A base technology program aimed at reestablishing the capability to manufacture and test fuel, with a follow-on goal of improving the technology to the point where it can support deployment of a VHTR economy, is the lowest risk approach to achieving the program goals.
- The peak fuel centerline temperature (1250°C) can support VHTR operation at least to the lower end of the desired core-exit helium coolant temperature range (850°C–1000°C).

- The fuel development and qualification activities outlined in this Technical Plan will be supported by adequate DOE funding.
- Activities related to the licensing of a fuel vendor's product by the NRC Office of Nuclear Reactor Regulation and meeting the NRC-RES mandate of Appendix B quality assurance and control are outside the scope of this program.
- Commercialization of fuel manufacturing technology, including qualification/licensing of vendor-supplied fuel, will proceed with at least 50% industry cost-sharing during 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 significant difficulties are encountered during the development, irradiation testing, and subsequent experiments. Difficulties could impact the completion schedule significantly.

2.3 OVERALL PROGRAM REQUIREMENTS

- Establish a fuel development and qualification program that will
  - address the generic issues previously identified by NRC staff members in their preapplication reviews of the MHTGR and the later pebble-bed modular reactor;
  - produce a fuel manufacturing specification adequate to fabricate fuel that meets the performance requirements of the reactor designer;
  - reestablish the domestic capability to manufacture high-quality coated-particle fuel using prior U.S. experience and German coating process data;
  - improve the understanding of the fabrication process, its impact on as-fabricated fuel properties, and in turn their impacts on irradiation performance;
  - reestablish the domestic capability to manufacture prismatic and/or pebble-bed fuel elements (consistent with VHTR design options);
  - produce or obtain coated-particle fuel samples as needed to support required testing;
  - complete the design and construction of reactor test rigs for domestic irradiation testing of coated-particle fuel;
  - develop and qualify coated-particle fuel through generation and presentation of statistically significant irradiation, postirradiation heatup testing, and PIE data under normal operational, transient, and accident conditions consistent with designer requirements;
  - demonstrate sufficient margin to failure for this fuel form under normal operational transient and accident conditions;
  - improve the understanding of fuel behavior and fission product transport, and use that understanding to improve the fuel performance and fission product transport models to accurately predict fuel behavior and fission product transport under normal operational transient and accident conditions;
  - develop pertinent technical information that supports the selection and implementation of the fabrication process used by the VHTR fuel vendor; and
  - develop pertinent technical information that can supplement the VHTR fuel vendor's own licensing/qualification data in the topical report supporting VHTR licensing.
- Implement this Technical Plan such that it supports both prismatic and pebble-type fuel designs. The effort dedicated to each design will be proportionate with its associated level of industry interest and commitment. Early stages of the program should support both designs by concentrating on the 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 fuel; document, as appropriate, the applicability of fuel fabrication process development to pebble-type fuel production.

- Implement this Technical Program Plan in accordance with the DOE quality assurance (QA) requirements specified in 10 CFR 830 "Nuclear Safety Management," Subpart A, "Quality Assurance Requirements" and in DOE Order 414.1A, "Quality Assurance." All activities that have direct input to the irradiation test specimen fabrication and irradiation campaigns will be conducted in accordance with the national consensus standard NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications," published by the American Society of Mechanical Engineers (ASME). Each participating organization shall prepare specific Quality Assurance Plans (QAPs) for its assigned scope of work and may prepare additional project-specific plans for individual work-breakdown-structure (WBS) elements as appropriate.

3. PROGRAM ELEMENTS

This section provides a summary description of the activities required to meet the high-level goals and requirements identified in Sect. 2. Program elements include fuel manufacture, fuel and materials irradiation, safety testing and PIE, fuel performance modeling, and fission product transport and source term. The information presented here is based on a more detailed task breakdown provided in Appendices A–E.

3.1 FUEL MANUFACTURE

The ultimate goal for fuel manufacture is the economical production of ultrahigh-quality kernels, coated particles, and compacts. The fuel manufacturing activities described in this section will reestablish the capability to manufacture coated-particle fuel in the United States. Of necessity, the fuel manufacture effort must deal with both process optimization (to achieve the required kernel, coated-particle, and compact characteristics and quality) and production scale-up (where achieving needed throughput is not independent of the process).

Particle fuel fabrication is considerably different from historical power reactor fuel manufacturing. The fabrication process begins with UCO kernels that are formed by the internal gelation process in which droplets of a uranium-containing chemical broth are formed into gel spheres in a fluid medium. The resulting gel spheres are then stabilized, dried, and sintered into hard ceramic particles.

Fuel kernels are coated using fluidized-bed chemical vapor deposition processes. The coatings include a buffer carbon layer to hold fission products, an inner pyrolytic carbon (IPyC) layer to chemically protect the fuel kernel, a silicon carbide (SiC) layer that is equivalent to fuel cladding, and an outer pyrocarbon (OPyC) layer for external protection of the SiC. These coatings make each fuel particle into an independent sealed pressure vessel for control of fission products. The finished coated particle is a carbon and ceramic sphere that is stable to temperatures up to 1600°C. (See Fig. 3.1.)

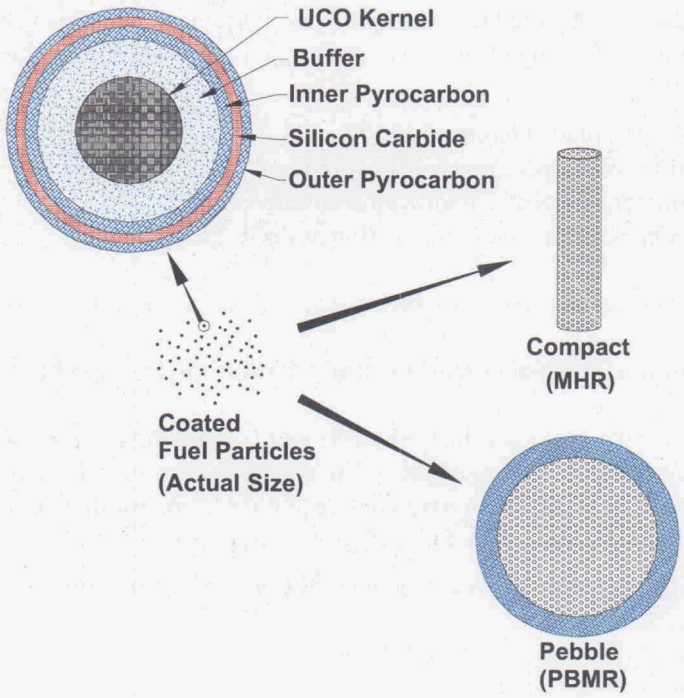

Fig. 3.1. Finished coated particle.

The coated fuel particles are agglomerated into physical shapes for use in the reactor. For the prismatic fuel design, fuel particles are compacted into a cylindrical rod shape for insertion in a large hexagonal graphite block suitable for loading into a reactor. For the pebble-bed design, fuel particles are compacted inside a graphite sphere, similar in size to a billiard ball, that is used in the reactor. For both designs, the particles are mixed with a thermosetting resin, formed into the desired shape, then carbonized and graphitized to provide a thermally stable material.

The target quality level for particle fuel will be that established in the German program in the late 1980s, with fuel element type GLE-4 (AVR reload 21-2) taken as a standard for comparison. The manufacturing effort will also seek to expand the understanding of the relationship between kernel and coating physical and mechanical properties and subsequent performance under irradiation. The earlier U.S. and German manufacturing efforts and the subsequent work in other national programs have achieved a substantial level of understanding, but additional work is still required.

Fuel failures in the MHTGR and NPR programs have been analyzed[4,5] along with U.S. and German fuel fabrication processes and irradiation performance.[6] These studies suggest key differences in German coating processes and properties that contribute to better irradiation performance. The most significant 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 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 is the good kernel and compacting experience in the United States, coupled with the successful German coating process as supplemented by the lessons learned from coating technology development in the United States.

The work to produce coated-particle fuel that meets the specifications includes kernel development, coating development, compacting development, scale-up analyses, 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 major R&D tasks required to establish particle fuel technology for the VHTR in the United States are discussed below. Schedule and cost information is provided in Sect. 4, with more detailed task lists, deliverables, and cost estimates in Appendix A.

The Fuel Manufacture Working Group has developed a fast track 5-year schedule, based on these aggressive assumptions:

1. unconstrained funding in place October 1, 2002;
2. immediate availability of key personnel and equipment;
3. management commitment from all organizations involved; and
4. very aggressive feedback from irradiation testing and PIE groups.

The activities needed to complete fuel fabrication process development are summarized below.

3.1.1 Prepare Irradiation Test Specifications and Irradiation Test Fuel Product Specifications

The starting point for the fuel manufacture effort involves preliminary specification of the basic reactor core requirements that the fuel must meet. These preliminary core design data are not part of the fuel manufacture effort but provide the prerequisite design envelope for development of a specific particle fuel. The information required includes the following:

- maximum operating and design basis accident condition fuel temperatures,
- maximum fast neutron fluence,
- maximum burnup expected out of the particle fuel,
- specific enrichment for fuel kernels,
- fertile kernels type (i.e., UO_2 or UCO),

- minimum and maximum fuel loading (packing fraction) for compacts, and
- determination of need or plan to use burnable poisons with fuel in compacts.

These data are to be provided by General Atomics (GA) and are based on previously completed Gas-Turbine Modular Helium Reactor (GT-MHR) design work, as documented in DOE-HTGR-10001 through -10248; they are augmented by earlier information from the modular high-temperature gas-cooled reactor (MHTGR) design and applicable information from the GT-MHR plutonium consumption reactor design. The information will be used for the generation of irradiation test specifications and irradiation test fuel product specifications to provide the requirements for the various irradiation tests and for the fuel samples to be included in these tests.

While the prismatic fuel design, product specification, and manufacturing information from the GT-MHR concept will provide the starting point for the VHTR fuel development and qualification effort, the program intends ultimately to develop and qualify fuel with a higher temperature capability. The SiC-TRISO concept will be retained for the immediate future while the program reestablishes the capability to manufacture and test fuel. The VHTR design, to be finalized later, will define the interfaces between the reactor plant and the process heat application that the reactor will support. It may be that the VHTR fuel design will evolve into a system similar to the SiC-TRISO system, but with the SiC replaced by a higher-temperature-resistant material. These details are not currently available to support planning.

3.1.2 Fuel Kernel Manufacturing

Babcock & Wilcox (B&W), now BWX Technologies (BWXT), used the reference internal gelation process to make large quantities of 200-μm kernels for the NPR program in the early 1990s. B&W also fabricated 350-μm kernels for the DOE commercial GT-MHR project in 1994. The main developmental efforts associated with the UCO kernel process are to optimize the process conditions for making 350-μm kernels and to qualify a substitute for trichloroethylene [(TCE) which requires special handling as a mixed waste] from the kernel forming process.

Initially, 5 kg of LEU UCO kernels and 5 kg of NUCO kernels will be fabricated for use in the small-coater work described in Sect. 3.1.3. Large composites of LEU UCO kernels (~40 kg) and perhaps NUCO kernels (~30 kg) will then be fabricated for use in the large-coater work and for fabrication of performance test fuel. Smaller composites of LEU UCO and perhaps NUCO kernels (~10 kg each) for qualification test fuel will be fabricated at a later date. Because of the previously noted uncertainties regarding the VHTR core and fuel designs, changes to this kernel procurement strategy are likely to occur once the missing information is provided by the VHTR program. Such possible changes include kernel diameter, enrichment, single vs multiple enrichment strategies, and UO_2 vs UCO unenriched kernels.

3.1.3 Coating Process Development

The lowest risk path to successful coated-particle manufacturing is to first "replicate" the proven German coating technology to the extent possible on a VHTR particle design, incorporating the lessons learned from prior U.S. fabrication and irradiation experience. The reference LEU UCO kernel for the VHTR is 350-μm UCO (rather than 500-μm UO_2 as used in the German fuel), so the German coating process parameters must be adjusted for the different kernel. "Replication" of the German coating process involves the following approach:

1. Use German coater design and operational experience to provide a baseline from which an existing U.S. production coater could be modified to provide a coating environment equivalent to the coating environment in the German production coater.

2. Use German coating process parameter information in conjunction with coating process information from previous U.S. and international programs to develop a reference set of process conditions.
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 if coatings with equivalent coating properties have been achieved.
5. Iterate on steps 3 and 4 until coatings having properties equivalent to the properties of the German coatings are achieved.

Economic coated-particle production is a function of the coater size. Achievement of overall program goals will require the successful use of large coaters, capable of coating the maximum load of kernels with each run. However, coating process optimization using a production-size coater would be very expensive and require handling of substantial quantities of enriched material. It is thought that German coating process conditions (i.e., temperature, coating gas concentration, and gas velocity across the particles) can be duplicated in any size coater. Therefore, this development effort is designed with the majority of coating development performed in a small-size coater that will allow a large number of coater runs at a relatively low cost.

The initial coating development effort will involve experimental work in a 2-in.-diam laboratory-scale coater to resolve questions concerning adjustment of German coating process conditions for coating of the smaller LEU UCO kernels and to develop a better fundamental understanding of the coating process. The key issues concerning adjustment of the German process parameters follow:

1. <u>IPyC coating permeability vs anisotropy</u>. The process conditions used by the Germans to deposit the IPyC coating result in a highly isotropic, but somewhat permeable IPyC coating. The permeability of the IPyC coating is apparently acceptable when coating German UO_2 kernels, but it may be a problem when coating UCO kernels because they are less dense and may have a larger surface area than UO_2 kernels. The supposition that UCO kernels may have a larger surface area than UO_2 kernels is based on limited data and is a question to be resolved by the program. The lower density and the (supposedly) higher surface area of UCO kernels relative to UO_2 kernels could make the UCO kernels more susceptible to attack by HCl gas during the SiC coating process, which could result in heavy metal dispersion into the buffer and IPyC coatings and in a higher level of as-manufactured SiC defects. Therefore, it must be determined if the IPyC deposited using the reference process conditions results in excessive fuel dispersion and/or SiC defects. Should this be the case, process conditions that yield IPyC coatings having both acceptable permeability and anisotropy must be established.

2. <u>Smaller size of LEU UCO kernels</u>. The optimum batch size and fluidization gas flow for the smaller kernels must be determined.

3. <u>Uninterrupted coating</u>. 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 (German fuel), the volume of the particles increases by about a factor of 6 during the coating sequence. For the 350-μm kernels (VHTR fuel), the volume of the particles will increase by about a factor of 11. The process condition adjustments necessary to accommodate this larger increase in particle volume during an uninterrupted coating sequence must be determined.

4. <u>Lower SiC deposition temperature</u>. The high SiC deposition temperatures used in the U.S. program produce large, columnar grains oriented in the direction of deposition. Reduction of the deposition temperature by 50°C results in smaller, equiaxed grains, which have shown better resistance to fission metal migration. The data are somewhat confounded by changes in other variables. Also, the lower SiC deposition temperature may reduce heavy metal dispersion during SiC coating that may explain the difference in German and US IPyC.

The small-coater work will also include development of a comprehensive coating process model, which can be used to predict accurately the impact of changes in process conditions on coated-particle properties and quality. This effort will include a series of coating runs with surrogate kernels, NUCO kernels, and LEU UCO kernels in which the process parameters are systematically changed in accordance with an experimental matrix that will be defined to yield the information needed to develop and verify the model. The fuel particles for the fuel compacts to be included in the "shakedown" irradiation capsule will be fabricated using the centerpoint process developed during the small coater work.

Figure 3.2 illustrates the logic for the laboratory-scale coater work. The laboratory-scale coater work will be judged successful when it has resolved the questions concerning the adjustments necessary to the

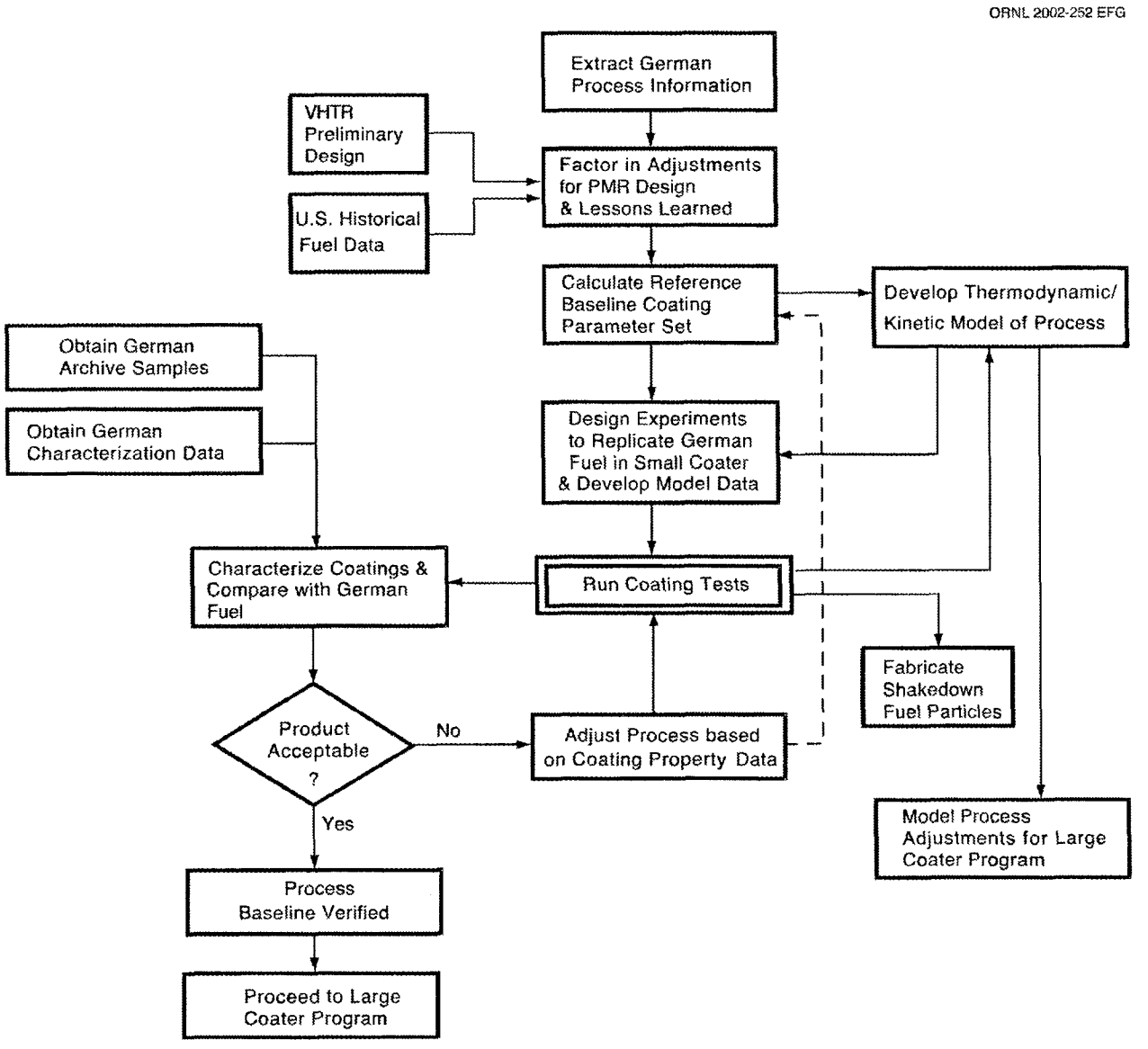

Fig. 3.2. Logic diagram for small-coater work.

German process conditions for 350-μm UCO kernels and has produced coated particles having coating properties representative of those in the reference German fuel particles. The laboratory-scale work is considered to be a cost-effective means of obtaining this information because the quantity of kernels required is small relative to the quantity of kernels required in a full-size coater.

The second phase of the coating process development effort will emphasize scale-up and manufacturing; thus the majority of the trials will be conducted using a large coater. Figure 3.3 illustrates the logic for the large-coater work. The Fuel Manufacture Working Group anticipates that a 6-in. coater currently installed and operational at BWXT will be appropriately modified and used for the large-scale coating work.

The initial work on the large coater will be limited to a few coating runs with surrogate or LEU UCO kernels to shake down the coater and verify that the reference set of coating conditions from small coater development extrapolates to the large coater. The evidence that this objective has been achieved will be obtained by characterizing the coatings to verify that they have the same desired properties as obtained in the laboratory-scale coater work. These coating runs will also be used to validate the coating process model.

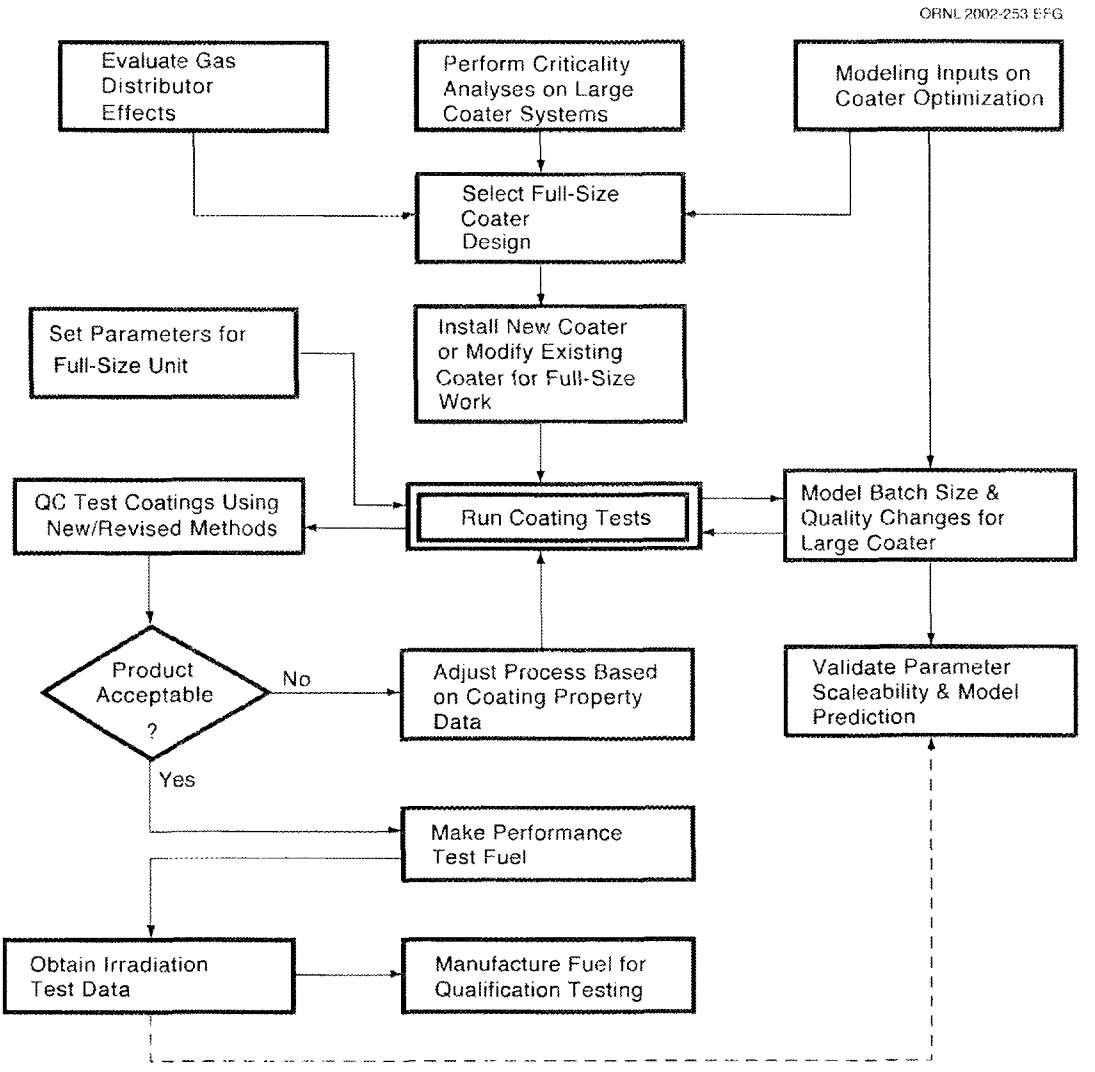

Fig. 3.3. Logic diagram for large-coater work.

Coating runs will be performed to fabricate the reference fuel particles and fuel particle variants for the fuel performance irradiation test. Coated particles that are expected to exhibit acceptable irradiation performance based on the similarity of their coating properties to the reference German particles will be made into compacts and irradiated in the fuel performance irradiation test. Coated particles fabricated using different coating conditions and having less similarity to the reference German particles will also be irradiated in the fuel performance irradiation test to assess the effects of the coating process variations on the irradiation performance of the fuel particles.

Cesium release data and accident condition performance data obtained from PIE and postirradiation heating of fuel must be evaluated to determine if the metallic fission product retention and accident condition performance of the reference fuel is acceptable. These data are needed to determine if any changes need to be made in the reference coating conditions used to fabricate the reference fuel particles.

Fuel will be manufactured for the fuel qualification test based on successful PIE and postirradiation heating of shakedown irradiation test material, completion of the fuel performance irradiation test, and indication of performance test success from fission gas release-to-birth ratio (R/B) data. Although the fuel manufacture effort will be proceeding with some financial risk to fabricate qualification test fuel with only R/B data from the performance test, the risk is considered acceptable. The PIE data on the performance irradiation test should be available before beginning the irradiation of the qualification fuel.

3.1.4 Quality Control Methods Development

The QC methods development effort includes a task to set up a QC laboratory that contains all equipment needed for basic QC inspection of coated particles and compacts, tasks to reestablish past capability, and, where appropriate, to develop new methods that are needed for enhanced characterization of kernels and coatings. New methods are needed for characterization of the stoichiometry of individual UCO kernels (i.e., the relative quantities of uranium oxide and uranium carbide), and IPyC coating permeability and anisotropy. Existing or previously used methods may be adequate to characterize SiC microstructure and defects. These tasks must be completed on an accelerated schedule in order for adequate characterization capability to be available to support the coating and compacting development work, and fabrication of fuel for irradiation testing.

3.1.5 Establish a Thermosetting Compacting Process

Compacting is a process for agglomerating fuel particles into a shape suitable for loading into a reactor. In the PBMR, a graphite sphere is used, and the fuel particles are agglomerated inside an outer spherical graphite shell. For the VHTR, cylindrical compacts of agglomerated particles would be fabricated and then loaded into a prismatic graphite block for installation in the reactor core.

Current U.S. compacting technology utilizes a thermoplastic matrix consisting of petroleum pitch mixed with graphite powder and injected into a mold to make compacts. The injection process requires pressurization of the particles, which is a potential particle failure mechanism. Also, the compacts must be supported by alumina powder during carbonization to prevent them from losing their shape. The raw materials used in the thermoplastic matrix have relatively high concentrations of metallic impurities that are highly reactive with SiC at high temperature, and the alumina powder used in the carbonization process is another source of impurities that can potentially attack the SiC coating.

For large-scale fuel manufacturing, a thermosetting-matrix-based process is preferred for a number of reasons. First, the thermosetting-matrix-based process would result in improved fuel quality because (1) the thermosetting matrix would be formulated from raw materials having lower levels of impurities; (2) the thermosetting matrix would yield stronger, less friable compacts; and (3) the thermosetting matrix process would involve lower compacting forces and less handling of the compacts, thereby reducing the potential for damage. Second, the thermosetting-matrix-based process would eliminate the need to pack the compacts in alumina powder during carbonization and would therefore be better suited to automation,

which would reduce the cost of fuel compact fabrication. Because of its many advantages, a thermosetting-matrix-based process will be developed to replace the current thermoplastic-matrix-based process as the reference compact fabrication process for the VHTR.

3.1.6 Fuel Product and Process Documentation

The initial fuel fabrication development and irradiation tasks are expected to provide the information to assist, along with additional reactor design information, in finalizing the top-tier fuel product specification that defines the requirements for fuel to be used in the VHTR. In addition to the fuel product specification, process modeling, test data, development results, and QC information will be thoroughly documented to define the processes necessary to successfully make and test fuel that will consistently meet the fuel product specifications and the performance requirements of the VHTR.

3.1.7 Technology to Mass-Produce Particle Fuel

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

- Develop automated QC methods that are adequate to support large-scale production. These methods should be nondestructive and capable of high throughput rates (potentially high enough to make 100% inspection feasible) and of providing near real-time feedback to the fuel fabrication processes.
- Develop a conceptual design for an automated fuel manufacturing plant.
- Prepare a waste minimization plan for future implementation in a fuel manufacturing plant.
- Perform a cost evaluation to develop estimates of the fuel unit cost for a commercially viable VHTR fuel manufacturing plant and the fuel unit cost potentially achievable in a fuel manufacturing plant utilizing the reference manufacturing processes. These cost estimates are needed to provide guidance to the pilot plant design effort and to identify process modifications that may need to be made to reduce manufacturing costs to a more acceptable level.

3.2 FUEL AND MATERIALS IRRADIATION

Irradiation testing of coated-particle fuels occurred routinely in the U.S. program from the 1960s through the early 1990s. Material test reactors have continued operation, and personnel who are experienced with all aspects of capsule design, assembly, and irradiation capsule monitoring remain in place at Oak Ridge National Laboratory (ORNL) and Idaho National Environmental and Engineering Laboratory (INEEL). Both laboratories were involved in irradiation testing of NPR-MHTGR fuel in the early 1990s. The ATR in Idaho and the High-Flux Isotope Reactor (HFIR) in Oak Ridge are both capable of irradiation testing of AGR fuels, although the HFIR cannot accommodate testing of full-size (60-mm) pebbles.

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

The fuel irradiation working group has developed a description of the tasks associated with irradiation testing of a representative capsule in the ATR and in HFIR. The following tasks were identified:

1. test specification;
2. capsule and supporting systems design, operational, and functional requirements;
3. capsule and supporting systems design;
4. capsule and supporting systems fabrication/assembly;

5. receipt of test articles;
6. review/approval of final design and fabrication data packages;
7. irradiation;
8. cooldown; and
9. disassembly and shipping.

While the details of capsule 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 detailed task listing, described in Appendix B (Sect. B.3) along with corresponding deliverables and interfaces with other activities, has served as the basis for schedule and cost estimates for the irradiation testing.

The number and type of capsules to be irradiated are dependent on the needs of the fuel manufacturer, fuel performance modeling, and fission product transport activities. Two types of capsules were identified for conducting the irradiations:

- multicell—a capsule containing six gas-tight independently controlled and monitored cells and irradiated in the large B hole of ATR (see Appendix B, Sect. B.1) and
- single cell—a single-cell capsule irradiated in the small B hole of ATR.

Irradiations in the single-cell capsule will require a shorter time because the fuel will be subjected to higher power and fast flux levels. The following capsules were identified based on discussions among the working groups during the course of developing the plan. Time and resource constraints limited the capsule conceptual design and performance information available to the fuel and materials irradiation working group. Further development of the capsule designs may alter the type of capsules to be used for individual irradiations. For example, it may be advisable to conduct the fuel qualification testing in single-cell capsules with a higher acceleration factor to shorten the schedule if more detailed information on the irradiation conditions shows this location to be acceptable.

3.2.1 Shakedown/Early Fuel (Capsule AGR-1)

This multicell capsule will include compacts made from early small-coater particles, possible compacts made from German particles, as well as possible unbonded particles and material samples. This will provide experience with multicell capsule design, fabrication, and operation, and it will reduce the chances of capsule or cell failures in subsequent capsules.* If successfully taken to a substantial fraction of design burnup and fast fluence, it will provide early data on irradiated fuel performance.† These early data on performance of fuel variants would support the development of a fundamental understanding of the relationship between the fuel fabrication process and fuel product properties and irradiation performance.

3.2.2 Performance Test Fuel (Capsule AGR-2)

This multicell capsule will include compacts containing particles made in a large coater from key variants in coated particles (e.g., IPyC permeability, anisotropy, uranium dispersion in buffer, and continuous vs interrupted coating). It may also include fuel performance modeling material samples (e.g., pyrocarbon and silicon carbide). To the maximum extent possible, common cell conditions, including

*If substantial early capsule failures (e.g., temperature control, gas monitoring, and instrumentation) are observed, the capsule could be removed and examined to determine required modifications in subsequent capsules.

†Particle variants tested should parallel those of subsequent large-coater product irradiations to the extent possible, potentially providing key data on metallic release and safety testing performance by the end of the first large-coater product irradiation (performance test fuel), in support of proceeding with fabrication of qualification test fuel as soon as possible.

burnup and temperature, would be maintained in the four central cells that contain the fuel variants to allow a direct comparison of performance. This capsule will provide irradiated fuel performance data and irradiated fuel samples for safety testing and PIE for key fuel product/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 and fuel product properties and irradiation performance.

3.2.3 Fission Product Transport (Capsule AGR-3)

Designed-to-fail (DTF) fuel particles for use in fission product transport testing will include particles with 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 multicell capsule will include compacts of unaltered LEU UCO and NUCO particles seeded with DTF fuel (i.e., DTF NUCO and LEU UCO particles seeded separately and located in separate cells, so the release from NUCO particles and LEU UCO particles can be determined separately). Unbonded DTF particles (LEU UCO and NUCO) will be irradiated in sealed capsules in fuel bodies. It will 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 product metal diffusion in kernels and gas and metal diffusion in coatings will be used in the development of input data for fission product transport models.

3.2.4 Fission Product Transport (Capsule AGR-4)

This multicell capsule will include DTF driver fuel and specialized samples of compact matrix and graphites. The capsule will be designed to provide data on fission product diffusivities and sorptivities in compact matrix and graphite materials for use in development of fission product transport models.

3.2.5 Fuel Qualification (Capsule AGR-5)

This multicell capsule 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* from AGR-1 and AGR-2. This will be the reference fuel design selected for qualification. Variations in cell conditions (burnup, fast fluence, and temperatures) will be established in accordance with the test specification. This capsule, in combination with capsule AGR-6, 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.

3.2.6 Fuel Qualification (Capsule AGR-6)

This multicell capsule will include the same fuel type as used in AGR-5. Variations in cell conditions (burnup, fast fluence, and temperatures) will be established in accordance with the test specification. This capsule, in combination with capsule AGR-5, 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.

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*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 some heat up and fission product metal release data on AGR-1 fuel, as well as in-pile gas release data from AGR-2.

3.2.7 Fuel Performance Model Validation (Capsule AGR-7)

This multicell capsule will include the same fuel type as used in AGR-5. The irradiation would cycle the fuel thermally and be designed so that some measurable level of fuel failure would occur. This capsule will provide irradiation fuel performance data and irradiated fuel samples for safety testing and PIE in sufficient quantity to validate the fuel performance codes and models and to demonstrate the capability of the fuel to withstand conditions beyond AGR-5 and -6 in support of plant design and licensing.

3.2.8 Fission Product Transport (Capsule AGR-8)

This multicell capsule will include compacts seeded with both NUCO and LEU UCO particles with missing buffers, unbonded reference particles, different temperatures among cells, and it will include temperature cycling. The capsule will provide irradiated fuel performance data and irradiated fuel samples for safety testing and PIE to determine material properties and fission product gas and metal releases from compacts with known quantities of failed particles for use in validation of fuel performance modeling and fission product transport codes.

3.3 SAFETY TESTING AND PIE

The purpose of this program element is to measure the performance of AGR fuel systems under normal operating transient and accident conditions. This work will support the fuel manufacture effort by providing feedback on the performance of kernels, coatings, and compacts. Data from PIE and accident testing will supplement the in-reactor measurements (primarily fission gas R/B) to answer the question of whether the fuel system is of sufficient quality to meet the reactor design requirements. And finally, data from this activity will form the basis of the licensee's fuel qualification submittal to the NRC to obtain an operating permit for the first plant.

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 the data needs. While most of the needed facilities and apparatus are currently in place, some modifications and upgrades will be necessary. In a few cases, the development of a new measurement technique may be required to satisfy a data need, which leads to an R&D task to develop or apply that new technique.

The HTGR fuel has been tested and examined at ORNL since the 1960s. The ORNL hot cells have a full range of capability to conduct the examinations that have been used in the past. In addition, the Argonne National Laboratory–West (ANL–W) hot cells (located near Idaho Falls, Idaho) have been used to examine a wide variety of irradiated fuels for many years. Both facilities are fully functional today. If additional examinations are required for the AGR program, both laboratories have competent development staff capable of designing and building the required equipment and developing the examination protocols.

The tasks associated with PIE and safety testing are briefly discussed below. As noted earlier, all PIE tasks may not be required. Determination of the required tasks for a particular capsule will occur during preparation of the PIE plan. Cost and schedule estimates for conducting PIE and safety testing are provided in Sect. 4. The detailed plan for this program element is located in Appendix C; it covers the full range of examinations performed in the past and comments briefly on additional examinations that may be required. 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 has few defective particles (i.e., a fraction of exposed uranium in the range of 10^{-5}), and in-pile R/B is low (i.e., $\sim 10^{-6}$), a lesser amount of PIE may be required, primarily addressing metallic fission product releases. Conversely,

if the as-manufactured fuel appears to be of high quality, but the in-pile R/B is unusually high (as was the case with the NPR fuel), then the PIE effort will be expanded to locate the source of the failures.

The following subsections discuss required preparations to conduct the PIE and safety testing activities, list the capsules with a brief summary of the PIE and safety testing objectives of each capsule, and then provide a detailed discussion of the PIE tasks with identification of the subset of tasks to be performed for each of the capsules, including cost and schedule estimates.

3.3.1 General PIE, Assessment, and Facility Preparation

The general PIE needs of the program involve capsule handling, capsule opening, fuel examination, fission product transport, fuel failure fraction determination, and accident testing. Much of the equipment required for these tasks is already in place, although some upgrading, improvements, and new capabilities may be necessary. Most of these tasks have been conducted in the past and present no unusual difficulties.

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 the capability of current facilities. Thus, an early task will be to determine the best way, within cost and schedule constraints, to conduct the PIE within the DOE complex. In addition, known equipment deficiencies exist and will need to be addressed in order to fulfill the needs of this program. The following six tasks address these needs.

3.3.1.1 PIE site task

Review the capabilities of candidate facilities, existing and new, for performing the separate PIE tasks, developing new equipment to perform new tasks not possible at present, and determining how these facilities might be integrated. Consider the implications of transport and time delays that might impact analysis, cost, and schedule.

3.3.1.2 PIE preparation task

Prepare the PIE facility for the AGR PIE and safety testing. Inventory the PIE capability, and note equipment that needs to be developed outside of this task. Finally, prepare the necessary environmental, safety, and health (ES&H) 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. While much of the HTGR fuels examination apparatus has remained operable over the years, minimal maintenance has been performed on this equipment and little upgrading has been done. Specific tasks include the fabrication of jigs and fixtures, testing and replacement of machining tools, testing and inspection of HTGR-specific apparatus, upgrading software for the Core Conduction Cooldown Test Facility (CCCTF), checking CCCTF wiring and flow paths, preparation of ES&H documentation, and review of operating procedures.

Generally, the nominal PIE time for an irradiation capsule is approximately 1 year, assuming that facilities and personnel are available. However, the AGR capsule irradiation schedule and potential PIE needs from other HTGR-related programs will result in multiple capsules undergoing PIE at the same time. Thus, expansion of the PIE capabilities at one site or the sharing of PIE work at two sites will be necessary to handle the workload. This need is most pressing for complex, time-consuming, tasks like high-temperature annealing. Thus, construction of an additional CCCTF and possibly an irradiated microsphere gamma analysis (IMGA) will be necessary to handle the workload within the schedule. The costs for duplication of additional, special one-of-a-kind apparatus have not been included in these cost estimates, and the program should examine the need for additional PIE infrastructure costs because multiple capsules require PIE services in the same time frame.

Important equipment deficiencies exist in the DOE complex for performing the required AGR tasks. Some of this results from the scrapping or obsolescence of equipment developed in the past, and some results from equipment that needs to be developed specifically for this program. Four tasks have been developed to address this need.

3.3.1.3 PGA equipment development task

Develop a particle gas analyzer (PGA) to crush a particle at a specified temperature and analyze the released gases. A throughput of a least several particles per day is required.

3.3.1.4 Helium/air/steam CCCTF task

Develop the capability to work with air and steam ingress conditions at the temperatures of programmatic interest. A new fuel heating facility will be developed to extend the chemical environment capabilities beyond that of the current helium atmosphere furnace and to handle the increased workload of the AGR program.

3.3.1.5 Coating physical properties equipment development task

Develop tools to investigate irradiated coating physical properties, especially the structure and anisotropic nature of carbon. Some material properties of interest might be strength, density, microstructure, layer bonding, permeability, and elastic modulus.

3.3.1.6 Fuel element reactivation equipment development task

Develop a method to reactivate a fuel element at a specific temperature so that its R/B can be measured.

3.3.2 AGR-1: PIE Shakedown Early Fuel Capsule

The first capsule to undergo irradiation and PIE is AGR-1. The purpose of this capsule is to gain experience with multicell capsule design, fabrication, and operation and to reduce chances of capsule or cell failures in subsequent capsules. It also provides early data on irradiated fuel performance and supports development of a fundamental understanding of the relationship between fuel fabrication processes and fuel product properties and irradiation performance. The specific PIE and safety testing tasks anticipated to be performed on this capsule are identified in Table 3.1. The individual task scopes are summarized in Sect. 3.3.10 and discussed in more detail in Appendix C.

3.3.3 AGR-2: PIE Performance Test Capsule

The purpose of the AGR-2 PIE is to provide irradiated fuel performance data beyond the R/B measurements for key fuel product/process variants to broaden options and increase prospects for meeting fuel performance requirements and to support development of a fundamental understanding of the relationship between fuel fabrication processes and fuel product properties and irradiation performance. The specific PIE tasks and safety testing tasks anticipated to be performed on this capsule are identified in Table 3.1.

3.3.4 AGR-3: PIE Fission Product Transport

The purpose of the AGR-3 PIE is to collect data on fission product metal diffusion in kernels and coatings for use in development of fission product transport models. This PIE will focus on specially designed transport specimens rather than the fuel. This PIE makes heavy use of the CCCTF, and additional capability in this area may be necessary. The specific PIE and safety testing tasks anticipated to be performed on this capsule are identified in Table 3.1.

3.3.5 AGR-4: PIE Fission Product Transport

The purpose of this PIE is to collect data on fission product diffusivities and sorptivities in graphite materials for use in the development of fission product transport models and overall codes. This PIE will focus on specially designed transport specimens rather than the fuel. The specific PIE and safety testing tasks anticipated to be performed on this capsule are identified in Table 3.1.

3.3.6 AGR-5: PIE Fuel Qualification

The purpose of this PIE is to collect irradiation 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. The primary interest is verifying proper fuel performance. This PIE also makes heavy use of the CCCTF. The specific PIE and safety testing tasks anticipated to be performed on this capsule are identified in Table 3.1.

3.3.7 AGR-6: PIE Fuel Qualification

The purpose of this PIE is also to collect irradiation 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. The primary interest is verifying proper fuel performance. This PIE also makes heavy use of the CCCTF. The specific PIE and safety testing tasks anticipated to be performed on this capsule are identified in Table 3.1.

3.3.8 AGR-7: PIE Fuel Performance Limits Capsule

The purpose of this PIE is to collect data on the capability of the selected fuel to withstand irradiation and accidents conditions beyond the conditions in capsules AGR-5 and -6 in support of plant design and licensing. The specific PIE and safety testing tasks anticipated to be performed on this capsule are identified in Table 3.1.

3.3.9 AGR-8: PIE Fission Product Transport

The purpose of this PIE is to collect irradiated fuel performance data on fission product metal releases from compacts with known quantities of failed particles for use in validation of fission product transport codes. This PIE also makes heavy use of the CCCTF. The specific PIE and safety testing tasks anticipated to be performed on this capsule are identified in Table 3.1.

3.3.10 Coating Diffusivity

The purpose of this PIE is to conduct annealing of deconsolidated particles from one or more of the above capsules to support the task outlined in "Fission Product Transport and Source Term," Sect. 3.5.3 (Appendix E).

Table 3.1. Capsule PIE tasks

Task No.	Task name	AGR-1	AGR-2	AGR-3	AGR-4	AGR-5	AGR-6	AGR-7	AGR-8
PIE TASK-1	Load Irradiation Capsule	X	X	X	X	X	X	X	X
PIE TASK-2	Capsule Gamma-Scanning	X	X	X	X	X	X	X	X
PIE TASK-3	Capsule Opening	X	X	X	X	X	X	X	X
PIE TASK-4	Component Metrology	X	X	X	X	X	X	X	X
PIE TASK-5	Fuel Element Cross Section	X	X	X		X	X	X	X
PIE TASK-6	Fuel Element R/B Reactivation		X	X				X	X
PIE TASK-7	Component Activity	X	X	X	X	X	X	X	X
PIE TASK-8	Leach-Burn-Leach	X	X			X	X	X	
PIE TASK-9	Fuel Element Deconsolidation	X	X					X	
PIE TASK-10	IMGA	X	X					X	
PIE TASK-11	Fuel Metallography	X	X			X	X	X	X
PIE TASK-12	Fuel Particle SEM Failure Mechanism	X	X					X	
PIE TASK-13	SEM Examination of Fission Products in Kernels and Coatings	X	X					X	
PIE TASK-14	Fission Gas and CO/CO_2 Content of Particle		X						
PIE TASK-15	Properties of Irradiated Materials Specimens	X	X						
PIE TASK-16	Radionuclide Transport in Irradiated Specimens		X		X				X
PIE TASK-17	Fission Product Release During Postirradiation Annealing	X		X	X	X	X	X	X
PIE TASK-18	Postannealing Metallography	X	X			X	X	X	
PIE TASK-19	Postannealing SEM	X	X			X	X	X	
PIE TASK-20	Waste Handling	X	X	X	X	X	X	X	X
PIE TASK-21	Reporting	X	X	X	X	X	X	X	X

3.3.11 Detailed PIE Scope of Activities

A capsule PIE 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 serially. For example, a capsule must be opened before any work can be done with the fuel, so it is a serial task. Fuel element 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, unrelated tasks. 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 a PIE will take approximately 1 year to complete with no restrictions on resources. The following tasks outline the options that are likely to be available for a particular PIE. The actual tasks that will be performed for a particular capsule are shown in Table 3.1.

The PIE tasks will have to be integrated with the activities of other AGR working groups so that the tasks can be conducted in an efficient and time effective manner. The primary goal is to ensure that the needed 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 capsules.

PIE TASK-1: Load Irradiation Capsule: Complete the transfer and nuclear accountability documentation, and prepare the hot cell for the delivery of the cask.

PIE TASK-2: Capsule Gamma-Scanning: Prepare the capsule for gamma scanning, and gamma-scan the capsule. Produce a false color image of the capsule and any regions that appear abnormal.

PIE TASK-3: Capsule Opening: Using in-cell machine tools and jigs, open the irradiation capsule, and remove the fuel elements and internal components of experimental value.

PIE TASK-4: Component Metrology: Visually and dimensionally inspect the fuel elements and capsule internal components.

PIE TASK-5: Fuel Element Cross-Section: Examine cross sections of a fuel element by optical metallography to document conditions within the fuel element, including fuel particles and matrix. The examination will visually document conditions within fuel particles such as kernel migration, kernel porosity, buffer integrity, and the integrity of the TRISO layers (i.e., IPyC, OpyC, and SiC).

PIE TASK-6: Fuel Element R/B Reactivation: Place fuel elements, one at a time, in a TRIGA or TRIGA-like reactor with an internal temperature-controlled furnace. This task will allow the individual measurement of fuel element R/B (rather the capsule total R/B) and the identification of fuel elements with damaged fuel particles.

PIE TASK-7: Component Activity: Individually gamma-count capsule components to determine the isotopes and amount of fission products present.

PIE TASK-8: Leach-Burn-Leach: Measure fuel particle failure fraction in irradiated fuel elements using the leach-burn-leach technique.

PIE TASK-9: Fuel Element Deconsolidation: Deconsolidate fuel elements by an electrochemical technique to obtain individual fuel particles; sieve particles to remove debris, and wash and dry.

PIE TASK-10: Irradiated Microsphere Gamma Analysis (IMGA): Gamma-scan a statistically significant number of particles to determine their fission product inventory, and identify and collect failed fuel particles by the IMGA technique.

PIE TASK-11: Fuel Metallography: Examine both good and failed fuel particles to document failures in TRISO coatings using optical metallography.

PIE TASK-12: Fuel Particle SEM Failure Mechanism: Examine failed fuel particles with a scanning electron microscope (SEM)/microprobe using wavelength dispersive X-ray spectroscopy (WDX) to elucidate the failure mechanism and map the isotopes of interest.

PIE TASK-13: Examination of Fission Products in Kernels and Coatings: Examine with an SEM/microprobe (using WDX) the components of intact TRISO fuel particles to measure fission product contents (mapping) and concentration gradients within the kernel, buffer, IPyC, SiC, and OPyC.

PIE TASK-14: Fission Gas and CO/CO_2 Content of Particle: Measure fission gas, CO_2, and CO contents of intact irradiated particles by mechanically breaking particles and collecting and analyzing the gases released.

PIE TASK-15: Properties of Irradiated Materials Specimens: Measure properties (thermal, physical, mechanical) as requested on samples of irradiated materials such as PyC, SiC, graphite, and metals.

PIE TASK-16: Radionuclide Transport in Irradiated Specimens: Measure radionuclide content and gradients in irradiated specimens by appropriate established techniques, such as beta and gamma spectrometry, and neutron activation.

PIE TASK-17: Fission Product Release During Postirradiation Annealing: Conduct postirradiation isothermal annealing tests to measure fission product release as a function of time at temperatures in the range of 1400°C–1800°C, as requested by the AGR program. These safety tests can be performed on fuel elements or unbonded fuel particles. Three atmospheres are to be considered as possible testing options: helium, helium/air, and helium/steam.

PIE TASK-18: Postannealing Metallography: Characterize TRISO-coating layer integrity by visual metallography, looking for evidence of SiC layer thinning and decomposition, chemical attack of SiC, and mechanical condition and microstructures of the SiC and PyC layers.

PIE TASK-19: Postannealing SEM: Measure (map) fission product distribution (especially Pd, Ag, and Cs) in fuel particles (kernels, buffer, coating layers) and fuel elements (graphite matrix) with an SEM/microprobe (WDX), looking for evidence of fission product accumulations at the IPyC/SiC interface, fission product attack of SiC, and fission products outside the fuel particles.

PIE TASK-20: Waste Handling: Collect, package, and dispose of wastes and spent fuel generated during the conduct of the AGR PIE.

PIE TASK-21: Reporting: Disseminate the findings, results, and problems of the PIE task in both formal and informal reporting. Support the program requests for specific information, clarifications, and impact assessments.

Costs and schedules for each capsule are provided in Sect. 4 and detailed in Appendix C.

3.4 FUEL PERFORMANCE MODELING

As discussed here, fuel performance modeling addresses the structural, thermal, and chemical process that can lead to coated-particle failures. It does not address the release of fission products from the fuel particle, although it considers the effect of fission product chemical interactions with the coatings, which can lead to degradation of the coated-particle properties. Many groups have attempted to model the performance of coated-particle fuels.[7] 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.

The current effort by the modeling working group is focused on improvements in these crucial areas. While the working group has provided a comprehensive identification of data needs, note that the schedules shown (see Sect. 4 and Appendix D) do not reflect the total time required to develop a model with the required accuracy and functionality. 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 IAEA Coordinated Research Project on fuel performance and fission product behavior.[8] 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 postirradiation heating, and (3) calculating fuel performance for AGR core designs during normal operation and hypothetical accidents. The development of fuel performance models requires a fundamental understanding of potential failure mechanisms and how these mechanisms depend on the irradiation conditions and the materials comprising the fuel. Accurate fuel performance modeling will also require good materials properties and constitutive relations. Table 3.2 summarizes the key fuel failure mechanisms associated with TRISO-coated-particle fuel and how these mechanisms depend on reactor service conditions and the particle design and performance parameters. The following failure mechanisms under irradiation were considered: (a) pressure vessel failure, (b) cracking of IPyC, (c) IPyC partial debonding, (d) kernel migration, and (e) fission product attack. Under accident conditions, fission product attack, SiC thermal decomposition, an increase in SiC permeability/SiC degradation, oxidation of the SiC layer, and rapid energy deposition were considered. Table 3.3 summarizes the important material properties that are 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. In addition, comments are made concerning the fact that Nuclear Energy Research Initiative (NERI) grants may cover much of this information. Nevertheless, they are retained in the list for completeness.

The scope of this section is limited to activities needed to support fuel performance modeling. However, as indicated in Table 3.2, 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 Working Group. 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 NERI and International NERI (INERI) projects.

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

3.4.1 Measurement of Anisotropy of PyC

Reliable, accurate, and precise methods are needed to characterize the anisotropy of PyC following fabrication and following irradiation. Existing data suggest that many of the key PyC material properties are a function of the anisotropy. Better methods, compared to the historical X-ray and optical techniques, are needed both for QA/QC and for fuel performance modeling.

Samples of PyC need to be fabricated and the anisotropy measured by both historical techniques and any new techniques. Both disks and coated inert particles are recommended for study. The samples also need to be irradiated and their anisotropy measured to determine if there is an influence of irradiation on anisotropy. Piggyback locations in the AGR-3 capsule are available to provide for the irradiation of these samples.

3.4.2 Thermomechanical and Thermophysical Properties of Coating Layers under Normal Operation

The thermomechanical and thermophysical properties of PyC and SiC listed in Table 3.3 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 material 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,

Table 3.2. 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 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	UO_2 only. Not important for UCO. Reasonably well understood
Diffusive release through intact layers	Temperature Burnup Temperature gradient Time at temperature	Chemical state/transport behavior of fission products Microstructure of SiC SIC thickness	Could be more important at high burnup in LEU fuels because of greater yield of noble metals (e.g., Ag) from plutonium fissions. More important during accident conditions
Fission product attack	Temperature Burnup Temperature gradient Time at temperature	Fission product transport behavior Diffusion? Buffer densification and cracking? 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

Table 3.2. (continued)

Failure mechanism	Reactor service conditions	Particle design and performance parameters	Comments
Parameters that strongly influence the failure mechanism			
Corrosion of SiC by CO	Temperature Burnup Time at temperature	Kernel type (UO_2, UCO) IPyC performance	CO is generated in particles with UO_2 kernels. At elevated temperatures, CO can attack the SiC layer if the IPyC layer is porous or has failed.
SiC thermal decomposition	Temperature Time at temperature	SiC thickness Microstructure of SiC	Not important in traditional accident envelope (peak temperature <1600°C)
Increase in SiC permeability/SiC degradation	Burnup Temperature Fluence	Microstructure of SiC? 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 heatup
Oxidation of SiC layer	Partial pressure of oxygen Temperature Time at temperature	Thickness of SiC layer Microstructure of layer	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 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

Note: ? indicates a potential parameter.

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 funded under a NERI project.

3.4.3 Thermochemical Properties of Kernel under Normal Operation

The thermochemical properties of the kernel listed in Table 3.3 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 VHTR. In addition, CO release from UO_2 is needed at burnups in excess of 10% FIMA at relevant reactor temperatures (up to 1300°C). Finally, measurements of kernel swelling for both UO_2 and UCO are needed, especially at high burnup.

3.4.4 Thermomechanical and Thermophysical Properties of Coating Layers under Accident Conditions

Table 3.3 lists the properties needed to model the mechanical behavior of the coated particle under accident conditions. The thermal expansion coefficient and elastic modulus of PyC are needed as functions of fast fluence and temperature (1200°C–1800°C). Also needed are the corresponding properties of SiC. This work is currently funded under a NERI project.

3.4.5 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 is failed when ~50% of the layer has been attacked. This work is currently funded under an INERI project.

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). A technique to characterize this enhanced permeability is needed to accurately capture the effect in models. This would include better characterization of microstructural changes in SiC following irradiation. Piggyback samples in AGR-3 capsule will provide the irradiation locations for the SiC material needed in this task.

Kernel swelling data are needed under rapid energy deposition conditions. Kernel swelling and kernel coating mechanical interaction may be critical to predicting failure in reactivity transients. These data can be obtained as part of PIE following reactivity testing. It is unclear whether reactivity testing should be part of the DOE AGR program. The likelihood of rapid (i.e., super prompt critical) reactivity transients that could induce fuel failures is design specific as is the need for data addressing these conditions. Further input from plant designers in conjunction with their interactions with the NRC is needed to resolve this question.

3.4.6 Code Benchmarking and Improvement

There has been significant activity around the world (e.g., GA, INEEL, MIT, and France) to develop improved fuel performance codes under normal operating conditions. Thus, it can be assumed that the codes for normal operation are probably adequate, given the activity to update the component models for the various failure mechanisms. However, our preliminary assessment is that new accident condition codes are needed.

Table 3.3. Key material properties needed for fuel performance modeling

Property	Current state of knowledge	Importance in modeling	How to measure	Other comments
Irradiation performance				
PyC anisotropy	Known to be critical to characterize PyC behavior. Ability to measure it accurately and precisely is in question	All key properties are thought to depend on anisotropy	X-ray, laser Raman, optical (questioned), other NDE techniques (acoustic?, UV,IR? magnetic?)	Critical QC/QA measure of acceptable PyC; needs development
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	Specific NERI projects may provide these data
PyC irradiation-induced creep	Uncertain with a factor of 5 uncertainty 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)	Specific NERI projects may provide these data
Poisson's ratio in creep	Reasonably well known. Literature data range from 0.3 to 0.5. Best estimate is 0.4. Probably is a function of density. Unclear if it is a function of anisotropy.	Has modest effect on stress in PyC layer	Special specimens	Specific NERI projects may provide these data
Strength of PyC	Data vary significantly. Some data 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	Specific NERI projects may provide these data

Table 3.3. (continued)

Property	Current state of knowledge	Importance in modeling	How to measure	Other comments
Irradiation performance				
Strength of SiC	Data vary significantly. Need data as a function of density, neutron fluence, irradiation temperature, and microstructure (large grain vs small grain and columnar vs equiaxed). Microstructure is a function of deposition conditions. Data are available for Chinese SiC. German data suggest that irradiation can reduce strength. The United States has correlated a lot of data and concludes that 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 use axial compression of a cylindrical plug inside SiC cylindrical sample. Key issue is linkage of data to microstructure.	Specific NERI projects may provide these data
Interfacial bond strength between SiC and PyC	Very little is known. Historic value of ~50 MPa is used in calculations. Tends to agree reasonably well with values from SiC/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.	Specific NERI projects may provide these data
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 (1000 to 1300°C) would be useful.	Lower importance given uncertainty in other parameters	Density (density gradient column) measurements	Specific NERI projects may provide these data
Irradiation-induced SiC creep	Limited data at low fluence.	Unknown given poor state of knowledge.	Split ring or bend strength relaxation techniques	Specific NERI projects may provide these data

Table 3.3. (continued)

Property	Current state of knowledge	Importance in modeling	How to measure	Other comments
Irradiation performance				
Fission gas release from the kernel	Data on gas release are reasonably well known for UO_2. Little to no data on UC, especially at high burnup O.	Direct contributor to pressure in particle	Can be measured by crushing particles or online from "intentionally failed" particles	Would need loose irradiated 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	Would need loose irradiated UO_2 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	CTE is different in the two orientations in PyC and depends on the anistropy 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 size of sample adds to overall difficulty in measurement and ultimate uncertainty
Elastic modulus of PyC	Modulus is a function of anisotropy, fluence, density, and temperature. Little 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	Specific NERI projects may provide these data
Elastic modulus of SiC	Data from fusion program show a 10% drop at reactor-relevant temperatures and radiation doses. Little data above 1000°C.	Critical for potential reactivity events where large temperature gradients may develop with the fuel particle	Resonant ultrasound spectroscopy or nanoindentation	Specific NERI projects may provide these data
Thermal expansion coefficient of SiC	Limited data suggest expansion is constant between 900°C and 1300°C. No systematic dependence on coating temperature or neutron irradiation. The presence of free carbon in SiC can reduce CTE by 40%	Critical for potential reactivity events where large temperature gradients may develop with the fuel particle	Conventional techniques	Small size of sample adds to overall difficulty in measurement and ultimate uncertainty

Table 3.3. (continued)

Property	Current state of knowledge	Importance in modeling	How to measure	Other comments
Accident performance: long-term heating/air ingress/rapid reactivity transients				
Fission product interactions with layers and potential degradation of properties	Unknown influence at present	Unknown at present	TBD	MIT will examine this influence under French INERI on particle fuel modeling
Buffer survivability	Failure of the buffer appears to be important to whether fission products get to the IPyC/SiC interface	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	This effect needs to be studied with the performance model before a definitive direction on the need for this work can be made
Kernel swelling under rapid energy deposition	Little data available under rapid energy deposition conditions	Kernel swelling and kernel-coating mechanical interaction may be critical to predicting failure in rapid reactivity transients	Part of PIE following reactivity transient testing	Unclear whether reactivity safety testing will be in scope of DOE AGR program

Note: ? indicates a potential measurement technique.

Benchmarking of fuel performance codes is needed. Pretest predictions and posttest 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, fuel qualification, and fuel-performance-limit irradiation capsules will provide much of the separate effects data needed to improve the fuel performance models. This work is partially funded under an existing INERI project, but additional resources are needed because the INERI's duration (2 more years) is short relative to this program's duration.

3.4.7 Code Verification and Validation

The verification and validation (V&V) of fuel performance codes are required for licensing. An independent integral capsule irradiation (capsule 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 VHTR. The technical basis will be codified in design methods (computer models) validated by experimental data. The approach is to take credit for all fission product release barriers (i.e., kernels, coatings, graphite, primary coolant pressure boundary, and reactor building) in order to meet protective action guidelines at the exclusion area boundary (EAB) 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 10^{-7} for normal operation and about 10^{-6} for core heatup accidents. Such stringent limits on particle failure are impractical 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 the paragraphs which follow are designed to produce validated fission product release models that are accurate to within a factor of 4 for fission gas and a factor of 10 for fission metal. The 4X and 10X values have been used previously to guide the development of fission product behavior models that supported the GCR designs in place in the 1980s and 1990s. Part of the Fission Product Transport and Source Term effort described in this Plan will be to establish and validate the required linkage between the reactor design and the off-site fission product release parameters that the NRC will evaluate during the licensing process. This will ensure that the 4X and 10X factors are appropriate for current PMR and PBR designs. The following tasks have been identified, with a more detailed discussion and task breakdown provided in Appendix E and cost and schedule information provided in Sect. 4.

3.5.1 Fission Gas Release, Including Iodine and Tellurium, from Failed Fuel Particles

The current database under normal operating conditions consists primarily of TRIGA measurements on laser-failed particles, including UCO kernels. 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 required to develop and refine gas release models with uncertainties <4X at 95% confidence. No new experimental work on hydrolyzed fuel is recommended at this time given the lower priority of water ingress in a direct-cycle AGR.

R&D Task 3.5.1.1: Measure fission gas release (Kr, Xe, I, and Te) from exposed LEU/natural UCO kernels (compacts seeded with "DTF particles," which are reference UCO kernels with a 10- to 15-μm PyC seal coat) irradiated in capsule AGR-3 under near normal HTGR flux over a range of temperatures. Update appropriate models in the SURVEY code.

R&D Task 3.5.1.2: Measure fission gas, including iodine, from irradiated (capsule AGR-3), 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 code.

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 test capsules. There are German data for Cs, Sr, and Ag in UO_2 kernels of intact particles that were irradiated under near real-time conditions as well as limited laboratory data on cesium release from ThO_2 kernels. Data on metal release from 20% FIMA LEU UCO kernels may possibly be derived from HFR B1/Cell 1. Gaps in the data needed 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, which is used to calculate full-core metal fission product release under normal operating conditions, the TRAMP/COPAR code, which is used to calculate "hot spot" metal fission product transport under normal operating conditions (and used for capsule analysis), and the SORS code, which is 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 <10X at 95% confidence.

R&D Task 3.5.2.1: Measure and model fission metal release from LEU/natural UCO fuel kernels in failed and intact particles under near real-time irradiation in capsule AGR-3. The DTF particles irradiated in piggyback samples and in seeded fuel compacts will be used. The principal information on metal release will be from mass balances derived from radiochemical measurements of fission products transported to the irradiation capsule interior. Data from the PIE of HFR B1/Cell 1 will be analyzed to derive fission metal transport properties. Update the fission metal release correlations in the TRAFIC/COPAR and TRAMP/COPAR codes.

R&D Task 3.5.2.2: Measure and model fission metal release from LEU/natural UCO fuel kernels in failed and intact particles irradiated (in capsule AGR-3) 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 code.

3.5.3 Fission Product Effective Diffusivities in Particle Coatings

The fuel particle coatings, particularly the SiC coating, are the primary barriers to the release of 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 pyrocarbon 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 are dependent on the structures of the SiC and PyC coatings, which vary with the coating manufacturing process. Therefore, fission product diffusivities must 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, fluence, and as-manufactured coating attributes to reduce uncertainties to <10X at 95% confidence.

R&D Task 3.5.3.1: Measure diffusivities of Ag, Cs, and Sr in SiC and I, Te, Xe, and Kr in PyC as a function of temperature, fluence, and as-manufactured coating attributes. Unbonded, irradiated particles (capsule AGR-3) and particles deconsolidated from irradiated (capsules AGR-1 and AGR-2) fuel elements with low particle failure rates will be heated and diffusivities determined by measuring time signatures of fission product releases from the particles. Diffusivities of fission gases in PyC will be measured at very high temperatures where SiC is thermally degraded.

R&D Task 3.5.3.2: Update correlations for fission product diffusivities in coatings in the TRAFIC/COPAR, TRAMP/COPAR, and SORS codes.

3.5.4 Fission Product Diffusivities/Sorptivities in Graphite

Fuel element graphite can significantly attenuate the release of fission metals and preclude the release of actinides from the core during normal operation and during CCCD transients. The present correlations for fission metal diffusivities in core graphite are derived largely from laboratory measurements on unirradiated nuclear graphites and from 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 data are limited for cesium and strontium on irradiated graphite and irradiated compact matrix material. The available data indicate that 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. Gaps in the needed data are mainly in the area of irradiated graphite. In addition, large uncertainties exist in the correlations of fission metal transport in graphite because many of the apparent variables are not treated explicitly. Sufficient single-effects test data are needed to develop and refine diffusivity and sorptivity correlations with uncertainties <10X at 95% confidence.

R&D Task 3.5.4.1: Measure diffusivities and sorptivities of Cs, Sr, Ag, and Pu in a fuel-compact matrix and fuel element graphites as a function of temperature, fast fluence, and, as appropriate, coolant impurities, system pressure (for Ag), and the extent of graphite oxidation under normal operating and CCCD conditions. The irradiation will be carried out in capsule AGR-4.

R&D Task 3.5.4.2: Improve model for fission metal transport in graphite in the TRAFIC/COPAR, TRAMP/COPAR, and SORS codes.

3.5.5 Fission Gas Release Validation Data

The SURVEY and SORS codes require data on fission gas release for code validation that are independent of the data used to develop the codes. The validation should assure that the predictive methods are accurate to within 4X 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 irradiation capsules. The noble gas release from FSV at the end-of-life was overpredicted by about a factor of 2, where hydrolysis may have been less severe than in lab tests. The noble gas release from Peach Bottom Core 2 at the end-of-life was underpredicted 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 >10X higher as-manufactured coating defects than required for modern direct-cycle HTGRs. Fission gas release from irradiation capsules 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 from data used to develop the SURVEY and SORS codes.

R&D Task 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 capsule AGR-8 under near normal HTGR flux over a range of temperatures for validation of the SURVEY code.

R&D Task 3.5.5.2: Validate the SURVEY code for calculating fission gas release by performing pretest predictions and posttest calculations with the code and comparing the results with measurements.

R&D Task 3.5.5.3: Measure fission gas, including iodine, from irradiated (capsule AGR-8) reference fuel with known failure fraction under CCCD conditions for validation of the SORS code. The atmosphere for the CCCD conditions will be determined as the core design and accident analyses evolve.

R&D Task 3.5.5.4: Validate the SORS code for fission gas release by performing pretest predictions and posttest calculations with the code and comparing the results with the measurements.

3.5.6 Fission Metal Release Validation Data

The TRAFIC/COPAR and TRAMP/COPAR codes for calculating fission metal release under normal operating conditions and the SORS code 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 assure that the predictive methods are accurate to within 10X at 95% confidence. The validity of the codes for predicting fission metal release under normal operating conditions have been assessed by applying them to predict the observed metal release in operating HTGRs (Peach Bottom Core 2 and FSV) and in irradiation capsules and in-pile loops. Most of the available data are for cesium, with a small amount of silver and strontium data. In general, the releases of fission metals were underpredicted 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 assessed systematically.

R&D Task 3.5.6.1: Measure fission metal release from irradiation capsule AGR-8 contain LEU/natural UCO fuel with known failure under normal operating conditions for validation of the TRAFIC/COPAR and TRAMP/COPAR codes.

R&D Task 3.5.6.2: Validate the TRAFIC/COPAR and TRAMP/COPAR codes for fission metal release by performing pretest predictions and posttest calculations with the code and comparing the results with the measurements.

R&D Task 3.5.6.3: Measure fission metal release from irradiated (capsule AGR-8) reference fuel heated under CCCD conditions for validation of the SORS code. The atmosphere for the CCCD conditions will be determined as the core design and accident analyses evolve.

R&D Task 3.5.6.4: Validate the SORS code for fission metal release by performing pretest predictions and posttest calculations with the code and comparing the results with the measurements.

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 (O&M), especially for a direct-cycle plant. Correlations currently available that describe the deposition behavior of condensable radionuclides on structural metals have very large uncertainties (≫10X). 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 (tungsten). The current database is inadequate to estimate the potential importance of diffusion of deposited radionuclides into the interior of structural metals (indiffusion) at operating temperatures. Data are needed to characterize the deposition of Cs, Ag, I, and Te on structural metals. Correlations are needed which 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 <10X at 95% confidence.

R&D Task 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 Task 3.5.7.2: Update correlations for fission product deposition on structural metals in the PADLOC code.

3.5.8 Radionuclide Reentrainment Characteristics for 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 (≫10X). The liftoff database was obtained in blowdown tests wherein the test specimens were mechanically removed from the loop or reactor in which the plateout activity was originally deposited. These ex situ blowdown 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 blowdown to that during normal operation—and, to a lesser extent, the duration of the blowdown. No correlation between the fractional liftoff and the blowdown temperature or the humidity of the helium was evident. Ex situ liftoff data from the Japan Atomic Energy Research Institute (JAERI) (small contaminated samples cut from a component of the OGL-1 loop) suggest that reentrainment may be relatively modest even for very large shear ratios, SR > 100. High-quality liftoff data were obtained in COMEDIE BD-1; however, the materials of construction and service conditions were for the steam-cycle MHTGR, and the effect of dust was minimized by use of a full-flow filter (a planned second test with

dust was not performed). The extent to which plateout activity may be removed during rapid depressurization transients must be quantified for VHTR materials of construction and service conditions. Correlations are needed for I, Sr, Cs, Te, and Ag as a function of SR, wall shear stress, blowdown 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 <10X at 95% confidence.

R&D Task 3.5.8.1: Measure reentrainment of the key radionuclides just identified under the conditions specified in a series of out-of-pile loop tests.

R&D Task 3.5.8.2: Improve reentrainment model in POLO code.

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 (<10X at 95% confidence) for normal operating conditions and for CCCD transients. The data must be independent of those used to develop 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°C–500°C, well below the surface temperatures in the gas turbine. 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 these data have been 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 have not been analyzed. 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 turbine and recuperator materials of construction and need to be performed under VHTR service conditions; it is highly desirable to include a small, simulated turbine because there are no existing plateout data on rotating machinery. It is also desirable to include the effects of dust on the plateout distribution.

R&D Task 3.5.9.1: Measure radionuclide plateout from integral tests under normal operating and CCCD conditions in an in-pile loop. The effects of dust should be quantified. Two tests are planned: one under "clean" conditions and the other with dust added.

R&D Task 3.5.9.2: Validate the PADLOC code by performing pretest predictions and posttest calculations and comparing results with measurements. Analyze data from the COMEDIE SR-1 test and compare code calculations with data.

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 assure predictive accuracy within 10X at 95% confidence. The data must be independent of those used to develop 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 blowdown test of the CPL 2/4 in-pile loop, <0.5% liftoff of the deposited activity was observed; however, the maximum SR was only 1.08. Moreover, the CPL 2/4 loop contained an inordinate amount of metal oxide aerosol, so the data are likely to be biased high. Considerable additional liftoff data were generated by the PIE of 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 made at Peach Bottom, Dragon, and AVR. 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 INEEL, 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. There are also German data from measurements made in the AVR; however, the considerable dust in the AVR resulted primarily from mechanical attrition of the circulating fuel spheres and the applicability of these data to a prismatic VHTR is debatable. Limited semiquantitative 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 (>10X) the amount of liftoff observed in ex situ blowdown 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 assure that dust effects do not preclude validating design methods to predict fission product transport within the primary coolant circuit to within an accuracy of 10X at 95% confidence.

R&D Task 3.5.10.1: Measure radionuclide reentrainment (liftoff) from integral tests during rapid depressurization transients in an in-pile loop. The effects of dust should be quantified. Two tests are planned: one under "clean" conditions and the other with dust added.

R&D Task 3.5.10.2: Validate the POLO code by performing pretest predictions and posttest calculations with the codes and comparing the results with the measurements.

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* (PAG) dose limits at the EAB 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 made 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 light-water reactor (LWR) containment buildings (e.g., the DEMONA tests in Germany). These LWR data may be applicable to the VLPC, but parameters, such as aerosol particle size and concentration, and fission product chemical forms will need to be evaluated. 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. The extent to which LWR data on radionuclide transport, especially transport within containment buildings, are applicable to the VLPC must be determined.

R&D Task 3.5.11.1: Determine by thermodynamic code calculations or small-scale experiments, if necessary, the chemical species to be expected under CCCD conditions in the VLPC. Investigate the applicability of methods and data for modeling radionuclide transport within LWR containments to conditions within the VLPC under CCCD conditions. Utilize, to the extent possible, LWR methods and data to calculate radionuclide transport and deposition within the VLPC under CCCD conditions. If necessary, supplement these methods with the required separate effects measurements on a relatively small scale. The radionuclides of interest are I, Cs, Te, Sr, and Ag.

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 (DF) 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 Task 3.5.12.1: Conduct laboratory tests of the pressure relief train filter to measure DF for key radionuclides. Select several candidate filter mediums, and test over a range of bed depths for the expected range of blowdown stream conditions. The effects of temperature and coolant chemistry must be treated explicitly. The chemical composition of the key radionuclides (I, Sr, Cs, Te, and Ag) must also be determined. Revise as necessary the predictive model and use it to optimize the relief valve train filter design.

3.5.13 International Cooperation

A number of AGR R&D programs are ongoing or in the process being organized on an international basis. Cooperation with these international programs provides a potential for contributing to the resolution of a number of the R&D tasks identified above while saving program cost and schedule time.

R&D Task 3.5.13.1: Review international AGR R&D programs and facilities/capabilities, and recommend opportunities for cooperation to reduce program costs and/or schedule in the resolution of fission product transport and source term R&D tasks.

4. PROGRAM SCHEDULE AND COST

In the development of this plan, priority has been given to the early activities in support of near-term execution of the plan. Issues associated with longer term activities have been identified that will be need to be addressed during the early execution phase and factored into the overall planning. These issues are not expected to affect the basic logic of the plan, but they may affect the details of its execution. The program plan anticipates industrial participation for full execution. It is expected that the plan will be periodically updated to reflect additional knowledge and the results of ongoing and completed work.

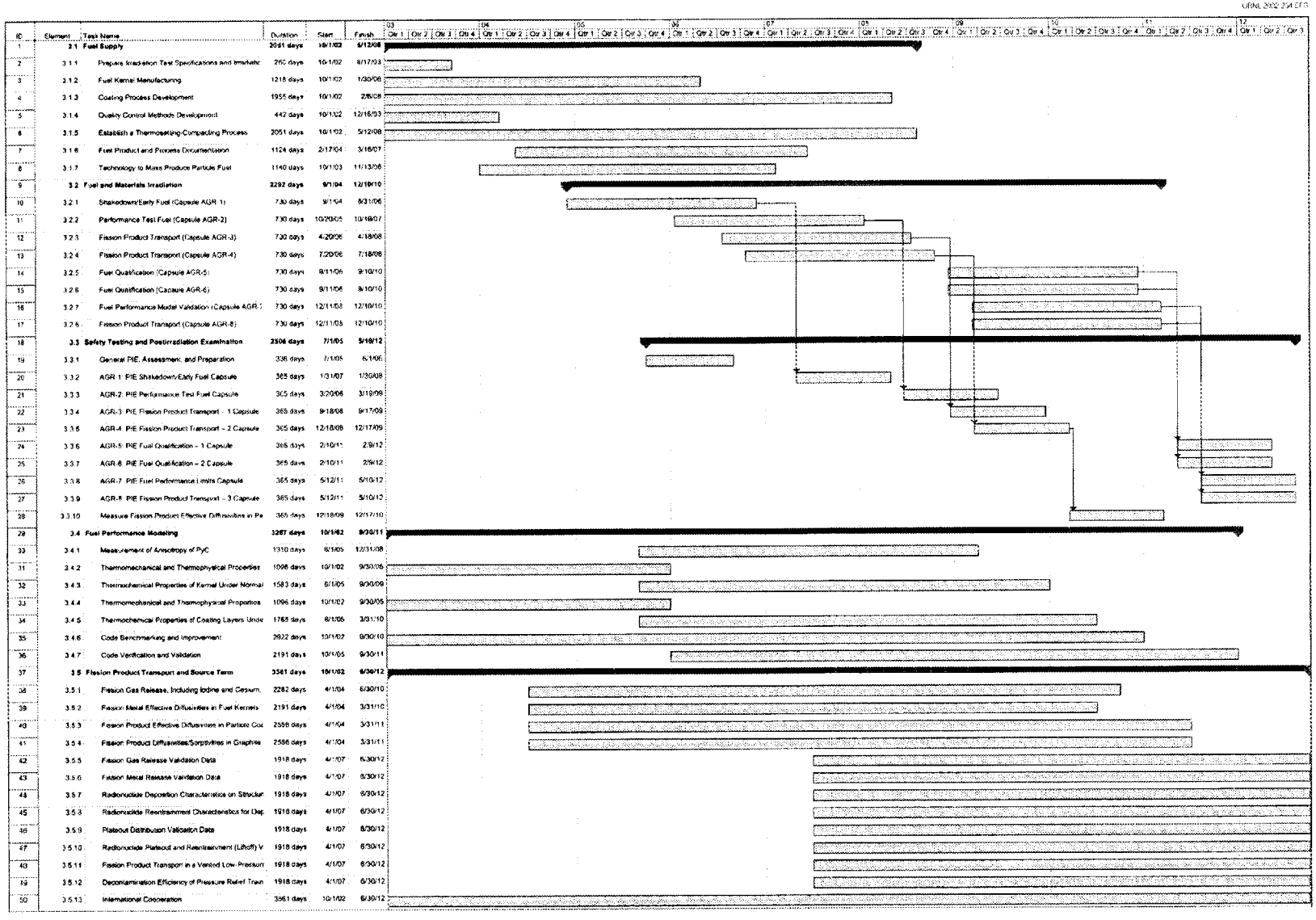
Detailed task breakdowns have been developed for each of the five program elements, along with cost and duration estimates, deliverables, and interfaces with the other program elements. A detailed discussion of the program elements and their corresponding tasks is provided in Sect. 3 and further expanded in Appendices A–E. These inputs have been used to develop an integrated program schedule and cost projection. The results shown in this section assume adequate funding (i.e., full funding) and a start date of October 2002 (beginning of FY 2003). The cost estimates at the program element level are presented below in Sect. 4.2.

4.1 SCHEDULE

Early program activities are centered on the fuel manufacture element because the production of fuel and materials for irradiation, safety testing, and PIE are the early critical path activities. Low 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. The approach taken to activity planning is success-oriented, but prudent in key areas. For example, as noted in Sect. 3.1.3, manufacture of fuel for the AGR-5 and -6 (qualification irradiation tests) is to be initiated based on the results of R/B data from the fuel performance test AGR-2, possibly coupled with limited PIE and safety testing data from the shakedown test AGR-1. If the in-pile performance of one or more of the fuel variants in AGR-2 is judged to be acceptable, the qualification fuel for AGR-5 and -6 will be specified and produced at risk, and the irradiation will proceed pending early PIE data from AGR-2. If the in-pile performance of all of the fuel variants in the AGR-2 test is unacceptable, time will be required to perform detailed PIE and feed back the learning to the fuel fabrication activity to correct the performance deficiency. This could easily result in a 2- to 3-year delay as new performance fuel fabrication and testing work is performed. Similar delays could be encountered if in-pile behavior of one or more performance test fuel variants is acceptable, but the later PIE or safety test data from the selected fuel indicate unacceptable behavior.

The approach taken to the scheduling of PIE and safety testing activities is to perform the work as quickly as possible. For the purposes of this plan, the PIE and safety testing activities are performed concurrently, and each has a duration of 1 year. In the schedule, this leads to an overburdening of the available PIE facilities and the obvious need to load level during some periods, particularly the time frame following the AGR-5, -6, -7, and -8 irradiation tests. The program planners' decision at this time is that it is not prudent to expend effort generating detailed plans beyond FY 2009. As time progresses and real information is generated by the work, task planning (budget and resource load leveling) will occur in response to results, budget allocation, and learning. This is not to say that issues are ignored. Early PIE tasks include facility evaluation and work allocation within the DOE complex, as well as planning for out-year equipment needs to mitigate known bottlenecks (e.g., design and fabrication of new equipment for work flow limiting facilities such as the CCCTF). Further, it may be that by FY 2009 licensing requirements will drive safety testing tasks more strongly than presently recognized decision points in the program plan. Also, the need to have the gas reactor fuel qualified earlier for an FY 2015 VHTR demonstration plant deployment may require increases in funding and earlier task scheduling.

Given these assumptions and disclaimers, the integrated schedule, taken to the first level below the program element level, is shown in Fig. 4.1.



ID	Element	Task Name	Duration	Start	Finish
1	3.1	**Fuel Supply**	2051 days	10/1/02	6/12/08
2	3.1.1	Prepare Irradiation Test Specifications and Irradiatio	260 days	10/1/02	8/17/03
3	3.1.2	Fuel Kernel Manufacturing	1218 days	10/1/02	1/30/06
4	3.1.3	Coating Process Development	1955 days	10/1/02	2/8/08
5	3.1.4	Quality Control Methods Development	442 days	10/1/02	12/16/03
6	3.1.5	Establish a Thermosetting-Compacting Process	2051 days	10/1/02	5/12/08
7	3.1.6	Fuel Product and Process Documentation	1124 days	2/17/04	3/16/07
8	3.1.7	Technology to Mass Produce Particle Fuel	1140 days	10/1/03	11/13/06
9	3.2	**Fuel and Materials Irradiation**	2292 days	9/1/04	12/10/10
10	3.2.1	Shakedown/Early Fuel (Capsule AGR-1)	730 days	9/1/04	8/31/06
11	3.2.2	Performance Test Fuel (Capsule AGR-2)	730 days	10/20/05	10/19/07
12	3.2.3	Fission Product Transport (Capsule AGR-3)	730 days	4/20/06	4/18/08
13	3.2.4	Fission Product Transport (Capsule AGR-4)	730 days	7/20/06	7/18/08
14	3.2.5	Fuel Qualification (Capsule AGR-5)	730 days	9/11/06	9/10/10
15	3.2.6	Fuel Qualification (Capsule AGR-6)	730 days	9/11/06	9/10/10
16	3.2.7	Fuel Performance Model Validation (Capsule AGR-7	730 days	12/11/08	12/10/10
17	3.2.8	Fission Product Transport (Capsule AGR-8)	730 days	12/11/08	12/10/10
18	3.3	**Safety Testing and Postirradiation Examination**	2506 days	7/1/05	5/10/12
19	3.3.1	General PIE, Assessment, and Preparation	336 days	7/1/05	6/1/06
20	3.3.2	AGR-1: PIE Shakedown/Early Fuel Capsule	365 days	1/31/07	1/30/08
21	3.3.3	AGR-2: PIE Performance Test Fuel Capsule	365 days	3/20/08	3/19/09
22	3.3.4	AGR-3: PIE Fission Product Transport - 1 Capsule	365 days	9/18/08	9/17/09
23	3.3.5	AGR-4: PIE Fission Product Transport - 2 Capsule	365 days	12/18/08	12/17/09
24	3.3.6	AGR-5: PIE Fuel Qualification - 1 Capsule	365 days	2/10/11	2/9/12
25	3.3.7	AGR-6: PIE Fuel Qualification - 2 Capsule	365 days	2/10/11	2/9/12
26	3.3.8	AGR-7: PIE Fuel Performance Limits Capsule	365 days	5/12/11	5/10/12
27	3.3.9	AGR-8: PIE Fission Product Transport - 3 Capsule	365 days	5/12/11	5/10/12
28	3.3.10	Measure Fission Product Effective Diffusivities in Pa	365 days	12/18/09	12/17/10
29	3.4	**Fuel Performance Modeling**	3287 days	10/1/02	9/30/11
30	3.4.1	Measurement of Anisotropy of PyC	1310 days	6/1/05	12/31/08
31	3.4.2	Thermomechanical and Thermophysical Properties	1096 days	10/1/02	9/30/05
32	3.4.3	Thermochemical Properties of Kernel Under Normal	1583 days	6/1/05	9/30/09
33	3.4.4	Thermomechanical and Thermophysical Properties	1096 days	10/1/02	9/30/05
34	3.4.5	Thermochemical Properties of Coating Layers Unde	1765 days	6/1/05	3/31/10
35	3.4.6	Code Benchmarking and Improvement	2922 days	10/1/02	9/30/10
36	3.4.7	Code Verification and Validation	2191 days	10/1/05	9/30/11
37	3.5	**Fission Product Transport and Source Term**	3561 days	10/1/02	6/30/12
38	3.5.1	Fission Gas Release, Including Iodine and Cesium,	2282 days	4/1/04	6/30/10
39	3.5.2	Fission Metal Effective Diffusivities in Fuel Kernels	2191 days	4/1/04	3/31/10
40	3.5.3	Fission Product Effective Diffusivities in Particle Coa	2556 days	4/1/04	3/31/11
41	3.5.4	Fission Product Diffusivities/Sorptivities in Graphite	2556 days	4/1/04	3/31/11
42	3.5.5	Fission Gas Release Validation Data	1918 days	4/1/07	6/30/12
43	3.5.6	Fission Metal Release Validation Data	1918 days	4/1/07	6/30/12
44	3.5.7	Radionuclide Deposition Characteristics on Structur	1918 days	4/1/07	6/30/12
45	3.5.8	Radionuclide Reentrainment Characteristics for Dep	1918 days	4/1/07	6/30/12
46	3.5.9	Plateout Distribution Validation Data	1918 days	4/1/07	6/30/12
47	3.5.10	Radionuclide Plateout and Reentrainment (Liftoff) V	1918 days	4/1/07	6/30/12
48	3.5.11	Fission Product Transport in a Vented Low-Pressur	1918 days	4/1/07	6/30/12
49	3.5.12	Decontamination Efficiency of Pressure Relief Train	1918 days	4/1/07	6/30/12
50	3.5.13	International Cooperation	3561 days	10/1/02	6/30/12

Fig. 4.1. Integrated program schedule.

Note that the duration for fuel performance modeling, fission product transport, and source term work extends for long periods of time consistent with the nature of the work as well as early planning for irradiation of materials samples and subsequent PIE to address data needs.

4.2 COST ESTIMATE

The cost estimate for this program is summarized in Table 4.1. The total program cost from 2003 through 2013 is $103M, based on a fully funded program with an October 2002 (beginning of FY 2003) start date. The cost distribution for each program element, broken down by major tasks, is shown in Tables 4.2–4.7.

For the fuel manufacture program element, the cost by fiscal year reflects its location on the critical path to all other program activities. One of the program's major goals is to reestablish the capability to fabricate and characterize coated-particle fuel. As discussed in Sect. 3.1, the scope includes UCO and UO_2 kernel manufacture, small-scale coating process development, and process scale-up to large-scale coaters, QC and compacting development, along with process documentation and mass production analyses to facilitate technology transfer to the industrial sector. Note that reestablishing fuel fabrication infrastructure will include training new personnel and developing updated processes and procedures as well as equipment and facilities. The estimates include costs for nuclear materials management and control (2%) and QA (to NQA-1, 6%).

Cost details for the fuel and materials irradiation work are shown in Tables 4.3 and 4.4. Table 4.3 lists the total cost per capsule. As an example of the cost associated with a typical capsule, cost details for AGR-1 from the first step of generating the irradiation test specification document (i.e., detailing the test articles, required capsule operating conditions, and required data) through the final step of disassembling and shipping the irradiated test articles are listed in Table 4.4.

The costs associated with the performance of PIE and safety testing work are summarized in Table 4.5. As noted earlier, all PIE tasks may not be required for a given capsule. Determination of the required tasks will occur during preparation of the PIE plan. The costs shown reflect preliminary best estimates of the PIE and safety testing work that may be required to support the goals of the various irradiation tests.

Table 4.6 provides a listing of the fuel performance modeling R&D tasks described in Sect. 3.4 and shows the estimated cost to perform the work. As noted in Table 4.1 the temporal distribution of cost follows the availability of data produced from the irradiation activities. Fuel performance modeling is an iterative task, and model benchmarking and improvement are annual activities consistent with the availability of new materials property data and fuel performance results from irradiation tests. Note that Tasks 3.4.2 and 3.4.4 leverage upon existing and potential NERI-funded projects.

The estimated costs for fission product transport and source term program element are summarized in Table 4.7. As noted in the temporal cost distribution of Table 4.1, the majority of the work is performed following the completion of the fission product focused capsules AGR-3 and -4.

Table 4.1. Cost summary by fiscal year

Program element	Program funding by fiscal year ($K)											
	2003	2004	2005	2006	2007	2008	2009	2010	2011	2012	2013–2017	Total
Program planning and management	700	1,000	1,000	1,000	850	1,000	1,000	750	800	300	800	9,200
Fuel manufacture	6,563	6,866	3,477	3,106	294	335	0	0	0	0	0	20,640
Fuel and materials irradiation	0	1,450	2,175	3,370	1,550	4,130	2,045	1,510	720	0	0	16,950
PIE and safety testing	100	1,450	3,389	1,284	4,005	1,313	5,560	1,566	7,433	4,468	0	30,566
Fuel performance modeling	700	550	1,250	1,525	1,250	2,175	1,500	1,010	40	0	0	10,000
Fission product transport and source term	185	465	600	1,710	2,265	2,400	1,365	910	775	835	4,425	15,935
Total	8,248	11,781	11,891	11,995	10,214	11,352	11,470	5,746	9,768	5,603	5,225	103,291

Table 4.2. Detail of fuel manufacture estimated cost breakdown

Work category	Cost ($K) by fiscal year						
	2003	2004	2005	2006	2007	2008	Total
Kernel manufacture	1,021	1,841	80	24	0	0	2,966
Coating development	1,348	1,565	392	0	0	130	3,434
Compacting development	670	221	86	0	0	76	1,053
QC development	3,093	244	0	0	0	0	3,337
Irradiation test and fuel product specification	432	43	36	41	102	43	697
Product/process documentation	0	81	24	44	116	86	351
Mass production analysis	0	2,870	2,859	2,997	77	0	8,802
Subtotal	6,563	6,866	3,477	3,106	294	335	20,640

Table 4.3. Cost by capsule

Irradiation test	Cost ($K)
AGR-1	2,725
AGR-2	2,275
AGR-3	2,275
AGR-4	2,275
AGR-5	1,975
AGR-6	1,475
AGR-7	1,975
AGR-8	1,975
Total	16,950

Table 4.4. Sample cost detail for irradiation test AGR-1

AGR-1 irradiation test	Cost ($K)
Test plan documentation	50
Development of capsule and supporting systems design, operational requirements, and functional requirements document	100
Capsule (new design) and supporting systems design, fabrication and assembly	1,750
Capsule fabrication/assembly	0
Receipt of test articles	0
Review and approval of design and fabrication data packages	25
Irradiation	700
Cooldown	0
Disassembly and shipping	100
Total	2,725

Table 4.5. PIE and safety testing cost summary

Task	Cost ($K)
Facilities assessment	128
PIE preparation	640
Equipment development for PIE —PGA equipment development —Helium/air/steam CCCTF design, fabrication, and shakedown —Coating physical properties equipment development —Fuel element reactivation equipment development	 710 5,225 445 1,300
AGR-1 PIE	1,780
AGR-2 PIE	2,625
AGR-3 PIE	2,725
AGR-4 PIE	2,030
AGR-5 PIE	2,965
AGR-6 PIE	2,965
AGR-7 PIE	3,520
AGR-8 PIE	2,450
Measure fission product diffusivities	1,058
Total	30,566

Table 4.6. Fuel performance modeling cost summary

Data need	Cost ($K)	Notes
3.4.1 Measurement of PyC anisotropy	1,350	Measure as-fabricated condition and changes resulting from irradiation
3.4.2 Thermomechanical and thermophysical properties of coating layers under normal operating conditions	300	SiC and PyC property data as a function of fast neutron exposure and coating deposition conditions. Most of the cost is covered under NERI—*$650K savings*
3.4.3 Thermochemical properties of the kernel under normal operating conditions	2,100	Property data as a function of burnup and operating temperature
3.4.4 Thermomechanical and thermophysical properties of coating layers under accident conditions	300	Temperature range of 1200°C to 1800°C. Most of the cost is covered under NERI—*$350K savings*
3.4.5 Thermochemical properties of coating layers under accident conditions	850	This task would interface with an existing INERI
3.4.6 Model benchmarking and improvement	3,100	For normal and accident fuel performance models
3.4.7 Code V&V	2,000	V&V activity for normal and accident fuel performance models
Total	10,000	

Table 4.7. Fission product behavior and source term cost summary

Task		Cost ($K)
3.5.1	Measure fission gas, iodine, and tellurium release from failed particles	650
3.5.2	Measure fission metal diffusivities in fuel kernels	560
3.5.3	Measure diffusivities of fission products (gaseous and metallic) in particle coatings	325
3.5.4	Measure fission product diffusivities and sorptivities in graphite	600
3.5.5	Measure fission gas release (Kr, Xe, I, and Te) from UCO to validate code	850
3.5.6	Measure fission metal release from UCO fuel under NOC to validate codes	800
3.5.7	Measure radionuclide deposition characteristics on structural metals	3,125
3.5.8	Measure radionuclide reentrainment characteristics for depressurization	3,125
3.5.9	Obtain plateout distribution validation data	2,425
3.5.10	Obtain radionuclide reentrainment (liftoff) validation data	2,325
3.5.11	Obtain fission product transport through calculation and experiment in VLPC	650
3.5.12	Measure decontamination efficiency of pressure relief train filter	450
3.5.13	Support international cooperation	50
	Total	15,935

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*Click on the "A-Z Index," then click on "Energy" and then on "National Energy Policy." The Gen-IV report is available at the same site, by clicking on "A-Z Index," then on "Nuclear Research," and on "Generation IV Nuclear Energy Systems."

Appendix A. DETAILED FUEL MANUFACTURE TASK LISTING

This appendix provides detailed task and cost information for the work needed to establish fuel fabrication technology in the United States for the Very High Temperature Reactor (VHTR) and to fabricate test fuel and material specimens needed for irradiation testing. The tasks described in this appendix are based on the goals, assumptions, and requirements listed below.

Goals

1. Establish coated-particle fuel fabrication technology in the United States for the VHTR that is capable of producing fuel at a quality level at least as good as produced by German fuel technology and on a schedule consistent with the initial VHTR plant deployment schedule.
2. Develop a fundamental understanding of the relationship between fuel fabrication process and fuel product properties and irradiation performance.
3. Develop, in conjunction with cosponsoring industrial partner during the later stages of the program, automated fuel fabrication technology suitable for mass production of coated-particle fuel at an acceptable cost.
4. Establish fuel process and product specifications that define all the requirements the fuel must meet to ensure acceptable performance at a peak centerline fuel temperature of 1250°C under normal operating conditions.
5. Develop and document the manufacturing processes required to meet the fuel process and product specifications mentioned in Goal 4.
6. Develop an understanding of modifications to the product specification necessary to support a higher peak centerline fuel temperature and the process refinements that will accompany the modified product specifications.

Assumptions

1. The VHTR core will be of prismatic design. Where possible, the technology developed by this program will be applicable to a pebble-bed core design as well.
2. The goal peak centerline fuel temperature (1250°C) can support VHTR operation at least to the lower end of the anticipated core-outlet helium coolant temperature range (850°C–1000°C).
3. The details of the VHTR core design are not yet available, so fuel developers do not know whether a single-particle system (i.e., enriched only) or a two-particle system (i.e., enriched plus depleted or natural) will be required. The SiC-TRISO data from this program apply to both the one-particle and two-particle concepts. At this time, the baseline SiC-TRISO fuel particle to be developed by this program will contain a 350-μm LEU UCO kernel.
4. Fuel fabrication process development will focus on VHTR low-enriched uranium (LEU) fuel; applicability of fuel fabrication process development to PBMR fuel production will be documented.
5. The capability to mass-produce the high-quality coated-particle fuel economically is a prerequisite for commercial viability of advanced gas reactors (AGRs).
6. The 10–17% enriched UO_2 particles qualified by the Germans for burnups of up to about 10% fissions per initial metal atom (FIMA) are not adequate for the higher burnups and more stringent service conditions (e.g., fluence and burnup) in the VHTR.

7. The reference fuel particle will be 19.9% ^{235}U LEU UCO. A mixture of LEU UCO and NUCO particles may be utilized in the VHTR. The exact mixture ratio will be determined as part of the VHTR core design. The NUO_2 kernels may be used in lieu of NUCO if kernel migration proves not to be a problem.
8. The reference process for fabrication of UCO kernels for the enriched VHTR fuel particles is the internal gelation process initially developed at Oak Ridge National Laboratory (ORNL) and further implemented at BWX Technologies (BWXT). Because of the existing equipment setup and recent UCO fabrication experience, it is assumed that the UCO kernels for coating and compacting development work and for irradiation test fuel will be fabricated at BWXT. Surrogate kernels and DUO_2 kernels for initial coating development work will be fabricated by ORNL.
9. Fuel particles having coating properties equivalent to those of German fuel particles from proof test composite EUO 2358-2365 will perform well in fuel compacts under VHTR irradiation conditions.
10. The lowest risk path to successful coated-particle manufacturing is to replicate the proven German coating technology to the extent possible on a VHTR design particle, which incorporates the lessons learned from prior U.S. fabrication and irradiation experience.
11. The reference process for fabrication of VHTR compacts will use a thermosetting matrix.
12. Archive fuel samples from German and U.S. programs currently stored at BWXT for KAPL will be made available to the AGR Program for use in process development and in quality control (QC) methods development.

Requirements

1. Reestablish and demonstrate coated-particle fuel fabrication capability in the United States by using existing equipment to conduct process development work and to fabricate fuel performance test fuel prior to construction and operation of a fuel manufacturing pilot line, including a full-size production coater.
2. Conduct fuel process studies to establish coating conditions that yield coating layers having the same microstructural properties as the coating layers on German fuel particles.
3. Replicate the German coating technology with a process suitable for large-scale fuel production. This will be accomplished by utilizing a coater that provides a coating environment equivalent to the coating environment in the German production coater and that has appropriate features (i.e., for loading, unloading, sampling material from the coater, and cleaning).
4. Develop improved QC methods as required to improve the characterization of the fuel and to support large-scale manufacturing of AGR fuel (e.g., VHTR and PBMR).
5. Fabricate fuel as needed for irradiation testing, including designed-to-fail (DTF) fuel for fission product transport tests. The test fuel shall meet the product requirements in the fuel product specifications for the irradiation tests, which will be prepared by the reactor designer based on the requirements for the individual irradiation tests.
6. Prepare a fuel product specification for large-scale VHTR fuel manufacturing that defines all requirements that the fuel must satisfy to ensure acceptable fuel performance under VHTR operating and accident conditions.
7. Support scale-up from laboratory-scale operation to production scale through sharing of data and close collaboration between technical staffs at ORNL and BWXT.
8. Develop automation technologies that can be applied to fabrication processes (opportunity for industrial partnership).

A.1 PREPARE IRRADIATION TEST SPECIFICATIONS AND IRRADIATION TEST FUEL PRODUCT SPECIFICATIONS

Irradiation test specifications and irradiation test fuel product specifications will be prepared to provide the requirements for the various irradiation tests and for the fuel samples to be included in these tests. A document summarizing the fuel functional requirements and performance targets will be prepared. Kernel, coated-particle, and compact specifications to meet those functional requirements and performance targets will subsequently be prepared. The irradiation test program intended to demonstrate and qualify fuel for the VHTR is described in Sect. 3.2 of the main body of this Technical Plan.

A.2 FUEL KERNEL MANUFACTURING

Babcock & Wilcox (B&W), now BWXT, used the reference internal gelation process to make large quantities of 200-μm kernels for the New Production Reactor (NPR) program in the early 1990s. B&W also fabricated 350-μm kernels for the DOE commercial GT-MHR project in 1994. The main developmental efforts associated with the UCO kernel process are to optimize the process conditions for making 350- and 500-μm kernels and to qualify a substitute for trichloroethylene (TCE) in the kernel forming process. The TCE-containing wastes are listed under the Resource Conservation and Recovery Act; treatment and disposal of mixed uranium-TCE contaminated wastes are expensive, so a substitute for TCE is desired.

Initially, 5 kg of LEU UCO kernels and 5 kg of NUCO kernels will be fabricated for use in the small-coater work described below. Large composites of LEU UCO kernels (~40 kg) and NUCO kernels (~30 kg) will then be fabricated for use in the large-coater work and for fabrication of performance test fuel. Smaller composites of NUCO and LEU UCO kernels (~10 kg each) for qualification test fuel will be fabricated at a later date. The kernel manufacturing tasks described below are those necessary to fabricate the LEU UCO and NUCO kernels at the BWXT facility. Changes to this kernel procurement strategy are likely to occur once the VHTR program develops core and fuel designs. Such possible changes include kernel diameter, enrichment, single vs multiple enrichment strategies, and UO_2 vs UCO unenriched kernels.

Kernel Process Development

The following kernel process development tasks will be completed prior to fabrication of the initial kernel composites for the coating development work.

- **Carbon dispersion.** Reestablish carbon dispersion operation for UCO production. Carbon must be dispersed in the acid-deficient uranyl nitrate (ADUN) during the original forming of the fuel kernels to achieve the desired final UCO composition. Achieving adequate mixing and wetting of the carbon with other components of the solution is essential to minimizing carbon segregation in the final UCO kernel.
- **Dispersion upgrade.** Upgrade the dispersion technique from a sonic disperser to a high shear-mixing pump.
- **Carbon wetting.** Establish processing parameters to ensure complete wetting of the carbon in the ADUN and an acceptable carbon particle size to minimize clogging of the nozzles in the forming system.
- **Process parameters for 350-μm UCO kernels.** Establish process parameters for reliable fabrication of 350-μm UCO kernels.

Fabricate Kernels

- **Acid-deficient uranyl nitrate.** Utilize the existing ADUN process for dissolution of a mixture of UNH crystals and U_3O_8 or UO_3 in nitric acid to obtain a liquid feedstock of the appropriate enrichment for either LEU UCO or NUCO kernel production.
- **Kernel forming.** The ADUN is premixed with the dispersed carbon, and the hexamethylene tetramine (HMTA) is premixed with urea. These two mixtures are then combined and quickly chilled to below 0°C. The chilled broth is vibrated into individual droplets of an appropriate size that drop into a heated bath (historically TCE). As the temperature of the spherical droplet rises, the HMTA decomposes to release ammonia that causes the uranyl nitrate to precipitate, forming a gel sphere. The spheres are then further stabilized by being washed in ammonium hydroxide before being air-dried.
- **Sinter fuel kernels in a vertical fluid bed sintering furnace.** The dried fuel kernels are initially calcined to remove all water of hydration, which leaves a product of pure UO_3 + C. The kernels are then sintered; first in hydrogen at 1600°C to reduce the UO_3 + C to obtain UO_2 + UC_2, then in a mixture of argon and CO at 1800°C to increase kernel density and adjust the carbon content to the exact composition desired. The final kernel chemical composition is UO_2 + UC_2, but the short-form terminology of UCO has been used historically to identify this kernel compound.
- **Finalize gas flow rates and compositions.** Finalize gas conditions to ensure a completely converted kernel (e.g., no free carbon) having high density and low surface area, in addition to the correct carbon-to-oxygen atomic ratio.
- **Adjust for particle uniformity.** Monitor and adjust processing conditions to ensure particle-to-particle and batch-to-batch uniformity (based on existing chemical and metallurgical evaluation techniques).
- **Sort and blend kernel lots.** Sintered kernels will be sorted, utilizing sieves to determine the actual size fractions, and tabled to remove irregularly shaped kernels. Acceptable kernels will be blended together to ensure a large uniform lot to feed the coating development studies and irradiation test fuel fabrication.

Develop TCE substitute. Develop environmentally benign alternatives to TCE as a forming media to be used in large-scale kernel manufacturing. This task will be completed prior to fabrication of the LEU UCO or NUCO kernels for qualification test fuel.

Facility cleanup. Clean up the facility, and dispose of waste.

A.3 COATING PROCESS DEVELOPMENT

The lowest risk path to successful coated-particle manufacturing is to "replicate" the proven German coating technology to the greatest extent possible on a VHTR particle design, which incorporates the lessons learned from prior U.S. fabrication and irradiation experience. The reference enriched kernel for the VHTR is 350-μm UCO (rather than 500-μm UO_2 as used in the German fuel), so the German coating process parameters must be adjusted for the different kernel. Replication of the German coating process involves the following approach:

1. Use German coater design and operational information to provide a baseline by which a U.S. production coater could be modified to provide a coating environment equivalent to the coating environment in the German production coater.
2. Use German coating process parameter information in conjunction with coating process information from previous U.S. and international programs to develop a reference set of process conditions.
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 if coatings with equivalent coating properties have been achieved.
5. Iterate on steps 3 and 4 until coatings have been achieved that have properties equivalent to the properties of the German coatings.

The initial coating development effort will involve experimental work in a 2-in.-diam laboratory-scale coater to resolve questions concerning adjustment of German coating process conditions for coating of the smaller enriched UCO kernels and to develop a better fundamental understanding of the coating process. The key issues concerning adjustment of the German process parameters follow:

1. IPyC coating permeability vs anisotropy. The process conditions used by the Germans to deposit the IPyC coating result in a highly isotropic, but somewhat permeable IPyC coating. The permeability of the IPyC coating is apparently acceptable when coating German UO_2 kernels, but it may be a problem when coating UCO kernels because they are less dense and may have a larger surface area than UO_2 kernels. The supposition that UCO kernels may have a larger surface area than UO_2 kernels is based on limited data and is a question to be resolved by the program. The lower density and the (supposedly) higher surface area of UCO kernels relative to UO_2 kernels could make the UCO kernels more susceptible to attack by HCl gas during the SiC coating process, which could result in heavy-metal dispersion into the buffer and IPyC coatings and to a higher level of as-manufactured SiC defects. Therefore, it must be determined if the IPyC deposited using the reference process conditions results in excessive fuel dispersion and/or SiC defects. Should this be the case, process conditions that yield IPyC coatings having both acceptable permeability and anisotropy must be established.
2. Smaller size of enriched UCO kernels. The optimum batch size and fluidization gas flow for the smaller kernels must be determined.
3. 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 about a factor of 6 during the coating sequence. For the 350-μm enriched UCO kernels, the volume of the particles will increase by about a factor of 11. The process condition adjustments necessary to accommodate this larger increase in particle volume during an uninterrupted coating sequence must be determined.
4. Lower SiC deposition temperature. The high SiC deposition temperatures used in the U.S. program produce large, columnar grains oriented in the direction of deposition. Reduction of the deposition temperature by 50°C results in smaller, equiaxed grains, which have shown better resistance to fission metal migration. The data are somewhat confounded by changes in other variables. Also, the lower SiC deposition temperature may reduce heavy metal dispersion during SiC coating that may explain the difference in German and US IPyC.

Logic and Sequence of Coating Work

The small-coater work will also include development of a comprehensive coating process model, which can be used to accurately predict the impact of changes in process conditions on coated-particle properties and quality. This effort will include a series of coating runs with surrogate kernels, NUCO kernels, and LEU UCO kernels in which the process parameters are systematically changed in accordance with an experimental matrix that will be defined to yield the information needed to develop and verify the model. It is expected that the fuel particles for the fuel compacts to be included in the "shakedown" irradiation capsule will be obtained from these experimental coating runs. The particles

tested in the shakedown capsule will be LEU UCO made using the optimum coating conditions developed from the work with UO_2 and UCO kernels.

Figure 3.1 illustrates the logic for the laboratory-scale coater work. The laboratory-scale coater work will be judged successful when it has resolved the questions concerning the adjustments necessary to the German process conditions for 350-μm UCO kernels and has produced coated particles having coating properties representative of those in the reference German fuel particles. The laboratory-scale work is considered to be a cost-effective means of obtaining this information because of the small quantities of kernels required relative to the quantities of kernels that would be required if the experimental coating work were to be conducted in a full-size coater.

The second phase of the coating process development effort will be conducted using a large coater. Figure 3.2 illustrates the logic for the large-coater work. The Fuel Manufacture Working Group anticipates that a 6-in. coater currently installed and operational at BWXT will be appropriately modified and then utilized for the large-scale coating work. The initial work on the large coater will be limited to relatively few coating runs with surrogate and/or NUCO kernels to shake down the coater and with LEU UCO kernels to verify that the reference set of coating conditions established in the laboratory-scale coater work extrapolate to the large coater. The evidence that this objective has been achieved will be obtained by characterizing the coatings to verify that they have the same desired properties as obtained in the laboratory-scale coater work. These coating runs will also be used to validate the coating process model.

Coating runs will be performed to fabricate the reference fuel particles and fuel particle variants for the fuel performance irradiation test. Coated particles that are expected to exhibit acceptable irradiation performance based on the similarity of their coating properties to the reference German particles will be made into compacts and irradiated in the fuel performance irradiation test. Coated particles fabricated using different coating conditions and having less similarity to the reference German particles will also be irradiated in the fuel performance irradiation test to assess the effect of the differences on the irradiation performance of the fuel particles.

Following completion of the fuel performance irradiation test, the performance of the fuel variants [as indicated by the fission gas release-to-birth (R/B) data for the various cells] will be analyzed to determine if any changes need to be made in the reference coating conditions used to fabricate the reference fuel particles. Cesium release data and accident condition performance data obtained from PIE and postirradiation heating, if performed, of fuel irradiated in the shakedown capsule will also be evaluated to determine if the metallic fission product retention and accident condition performance of the reference fuel is acceptable. If the performance of the reference fuel is determined to be acceptable, a single batch of reference fuel particles will be fabricated and inspected/tested for fuel qualification testing. Otherwise, modifications to the coated-particle design and/or process conditions will be necessary to improve the performance of the fuel.

A.4 SMALL-COATER WORK

Evaluation of German process and recent TRISO coating activities.

- **Determine processing conditions.** Establish the process conditions under which the reference German particles were fabricated, and review the more recent TRISO coating activities to benefit from coating advancements made by others.
- **Evaluate German processing data set.** Evaluate German processing data set, including particle "batch" properties, heating means, reactant species selection, gas flows and control systems, gas injection and distribution systems, temperature measurement technique(s), and other relevant processing variables and control systems.

- **Analyze process variables.** Analyze the relative gas flow concentrations, active coating gas fractions, temperatures, deposition rates, and gas velocities for the reference German system for given batch sizes.
- **Determine reference set of process conditions.** Determine a reference baseline set of processing conditions that mimic the German process for use in modeling and experimental studies.
- **Evaluate reference German particle characterization data.** Gather and review all available characterization data for the reference German fuel particles. This includes German QC data and data from additional testing performed in the United States during the NPR and previous DOE-NE commercial MHTGR programs. Determine any testing that should be performed to obtain additional information on these particles.

Coating process science and modeling.

- **Generate computerized model.** Generate a complete computerized process model of the coating operation that uses input variables, computation fluid dynamics, thermodynamics, and reaction kinetics to predict the changes in coated-particle properties as a function of coating parameter changes.
- **Experiment design.** Generate a statistical experiment design to quantify the effects of modifying coating process variables from the German baseline, and validate the process model.
- **Process control systems.** Review state-of-the-art in process control systems that could be used to monitor and control coating deposition variables such as temperature and gas flows.
- **Feedback mechanisms.** Examine process variables such as differential pressure or exhaust gas chemistry to be used as feedback mechanisms for monitoring and optimizing the process during operation.
- **Interrupted coating vs sequential, uninterrupted coating.** Analyze the feasibility of sequential, uninterrupted or continuous "batch" coating as compared to interrupted coating, of the reference VHTR fuel design using German coating process parameters. Analyze as a function of coater size and equipment constraints.

Modify the BWXT 6-in. coater if it is selected for the large-coater work.

- modify SiC coater to allow uninterrupted coating,
- install direct MTS injection system, and
- install additional instrumentation, as needed.

Coating tests.

- **IPyC permeability vs anisotropy.** Perform a coating run using the reference set of processing conditions. Test the coated particles for SiC defects and inspect for evidence of heavy-metal dispersion into the buffer and IPyC coating layers (i.e., the presence of either would be an indication that the UCO kernel has been attacked by HCl gas during SiC coating, which would indicate that the IPyC coating deposited by the reference coating conditions is too permeable). Perform additional runs, as necessary, to determine IPyC coating conditions that result in a sufficiently impermeable IPyC coating layer, and characterize the anisotropy of the coating layer to determine if it is acceptable.
- **Sequential, uninterrupted vs interrupted coating.** Perform coating runs to
  1. determine the influence of particle volume increase (factor of 11 for reference LEU UCO particles) on batch size and coating process parameters;
  2. determine the effect of loading and unloading techniques on coating quality;
  3. generate test particles for analysis of differences between particles with once-through deposition of all layers and those with buffer + IPyC, SiC, and OPyC deposited in separate runs;

4. examine the differences between interrupted coated and sequential, uninterrupted coated particles;
5. calculate the potential differences in residual stress between the two coating techniques;
6. evaluate the effect of bonding between the IPyC and SiC and its relationship to coating method; and
7. determine if there is a correlation between coating method and the presence of particle defects.

- **Parametric tests.** Perform coating tests in accordance with the statistical experiment design to support coating process model development. Parameters that may be evaluated include
  1. diluent gas flow rates/velocities;
  2. coating gas composition;
  3. coating gas concentration/velocities and their effects on deposition rate, microstructure, and anisotropy;
  4. coating process temperature;
  5. coating frit/cone configurations; and
  6. coater diameter.

Process model verification. Use coating test results to verify that the coating process model is predictive of the coating characteristics obtained using different coating process parameters.

Fuel particles for "shakedown" test capsule. Fuel particles for the shakedown capsule will be fabricated using LEU UCO kernels after successful completion of the development runs. Characterization of the fuel particles to be included in the shakedown capsule will include comparison with the reference German particles using both standard QC inspection and improved QC methods, as applicable.

Made-to-fail particles and other samples as needed for fission product transport and source term task irradiation tests. Special defective particles and/or coating layer samples will be generated as needed for specific irradiation tests.

A.5 LARGE-COATER WORK

Evaluate gas distributor design effects. Perform a literature review, and use the coating process model to evaluate the relationship between gas distributor design and fuel quality.

Perform a nuclear criticality analysis. Determine the optimum coater size and limitations on LEU UCO particle batch size from the standpoint of criticality safety.

Coater tests.

- **Set large-coater parameters.** Use the coating process model to set the reference process parameters for the large coater based on the reference process conditions established by the laboratory-scale coater work (i.e., basically, the German process conditions with adjustments necessary because of the smaller LEU UCO kernel). All large coater work is now performed using the new U.S. process parameters developed during the small coater work phase.
- **Calculate batch limits.** Use the engineering model to estimate the optimum and limiting batch sizes for LEU UCO.
- **Perform coating runs.** Perform a limited number of coating runs to
  1. shake down the coater,
  2. verify the coating parameters necessary to obtain the fuel particle quality and coating characteristics achieved in the small coater, and
  3. evaluate the quality of material produced using large- vs small-coating batch size, and verify the limiting batch size for the large coater(s).

Process model validation. Use results of small-coater work and large-coater work to validate the coating process model.

Fabricate and inspect/test fuel particles for fuel performance irradiation test. Coating runs will be performed to fabricate the reference fuel particles and fuel particle variants for the fuel performance irradiation test. For the purpose of estimating the fabrication effort, it is assumed that the irradiation test capsule will have six individually purged cells, with each cell capable of accommodating six fuel compacts. The test samples will provide valuable data concerning the relationships between process conditions, coating properties, and irradiation performance. The test samples will be extensively characterized and compared with the reference German fuel particles and with the reference fuel produced in the small coater. The comparison will use both standard QC inspection/tests and candidate improved QC methods. It is anticipated that the following fuel variants will be irradiated in this test.

1. reference fuel (2 cells)
2. IPyC anisotropy variant 1 (1 cell)
3. IPyC anisotropy variant 2 (1 cell)
4. SiC microstructure variant (1 cell)
5. interrupted coating sequence variant (1 cell)

Evaluate results from fuel performance irradiation test. Analyze the performance of the fuel variants irradiated in the fuel performance irradiation test (as indicated by the fission gas R/B data for the various cells) to determine if any changes need to be made in the reference coating conditions used to fabricate the reference fuel particles. Also, evaluate the cesium release data and accident condition performance data obtained from PIE and postirradiation heating, if performed, of fuel irradiated in the shakedown capsule to determine if the metallic fission product retention and accident condition performance of the reference fuel are acceptable.

Make fuel particles for fuel qualification tests. Fabricate and inspect/test LEU UCO for fuel qualification testing.

A.6 QC METHODS DEVELOPMENT

The QC methods development effort includes a task to set up a QC laboratory that contains all equipment needed for basic QC inspection of coated particles and compacts and includes tasks to develop new methods that are needed for enhanced characterization of kernels and coatings. New methods are needed for characterization of the stoichiometry of individual UCO kernels (i.e., the relative quantities of uranium oxide and uranium carbide), IPyC coating permeability and anisotropy, and SiC microstructure and defects. These tasks must be completed on an accelerated schedule in order for adequate characterization capability to be available to support the coating and compacting development work and fabrication of fuel for irradiation testing.

Install QC equipment for coated-particle and compact inspection/testing. Set up equipment for performing standard QC measurements on coated particles and compacts. Coating properties to be measured using these standard methods include the following:

- **Coated particles**
  - coating thickness and density
  - particle sphericity
  - defective or missing coating layers
- **Compacts**
  - uranium content and homogeneity
  - uranium contamination
  - SiC defects
  - fuel dispersion (into buffer and IPyC)
  - diameter, length, and integrity
  - impurities

Develop methods for enhanced characterization of kernels and coatings.

- **UCO kernels.** Develop a quantitative, nondestructive analytical technique for determination of the phase composition of individual kernels. Candidate methods include, but are not limited to, electrical resistivity measurements, Raman spectroscopy, and/or X-ray diffraction techniques.
- **Inner pyrocarbon permeability.** Develop a method for determining the effective impermeability of the IPyC coating to HCl at the SiC coating temperature. The HCl leaching technique previously explored at GA, which provides an indication of the permeability of the IPyC by measuring the amount of reaction between HCl and the kernel, is a candidate for this method.
- **Pyrocarbon microstructure/anisotropy.** Develop an improved method for characterizing the anisotropy of the IPyC coating. Candidate methods include, but are not limited to, BAF, Raman spectroscopy, and small-angle X-ray scattering (SAXS).
- **Silicon carbide defect identification and characterization**
- **SiC microstructure.** Develop a method for enhanced characterization of SiC microstructure.
- **SiC fission product retentiveness.** Develop an improved method of measuring the fission product retentiveness of the SiC coating. The method could either measure fission product retention directly or be a more sensitive method of measuring SiC through defects (i.e., defects that penetrate the SiC coating).
- **SiC defects.** Develop methods capable of detecting SiC defects (such as soot and impurity inclusions and poor grain structure) that do not penetrate the SiC coating, but that could lead to SiC failure during irradiation or accident conditions.

A.7 ESTABLISH A THERMOSETTING COMPACTING PROCESS

Current U.S. compacting technology utilizes a thermoplastic matrix consisting of petroleum pitch mixed with graphite powder and injected into a mold to make compacts. The injection process requires pressurization of the particles, which is a potential particle failure mechanism. Also, the compacts must be supported by alumina powder during carbonization to prevent them from losing their shape. The raw materials used in the thermoplastic matrix have relatively high concentrations of metallic impurities that are highly reactive with SiC at high temperature, and the alumina powder used in the carbonization process is another source of impurities that can potentially attack the SiC coating.

For large-scale fuel manufacturing, a thermosetting-matrix-based process is preferred for a number of reasons. First, the thermosetting-matrix-based process would result in improved fuel quality because the thermosetting matrix would be formulated from raw materials having lower levels of impurities; the thermosetting matrix would yield stronger, less friable compacts; and the thermosetting-matrix process would involve lower compacting forces and less handling of the compacts, thereby reducing the potential for damage. Second, the thermosetting-matrix-based process would eliminate the need to pack the compacts in alumina powder during carbonization and would therefore be better suited to automation, which would reduce the cost of fuel compact fabrication.

Because of its many advantages, a thermosetting-matrix-based process will be developed to replace the current thermoplastic-matrix-based process as the reference compact fabrication process for the VHTR. The development effort includes the following tasks.

Technology transfer. Pursue technology transfer through international collaboration (e.g., Japan, China, South Africa, Germany, etc.). Obtain processing parameters for use in the United States, or have U.S. fuel particles compacted by collaborating manufacturers.

Establish compact fabrication capability.

- **Evaluate potential source materials and resin-matrix compositions.** Commercially available, chemically pure, resins mixed with sized graphite powders will be evaluated for use in compacting. A variety of resins and resin/filler formulations will be tested to determine the optimum raw materials and formulation for compact forming, carbonization, and heat treatment.

- **Determine optimum conditions for compact formation.** The compacting conditions, such as premixing of particles and matrix and compacting pressure and temperature, will be optimized to produce compacts having the desired matrix density and uniformity of particle distribution without damaging fuel particles.
- **Develop carbonization process parameters.** The carbonization process must be carried out in an inert atmosphere or vacuum to prevent loss of the carbon as CO or CO_2. Also, the heating cycle must be carefully controlled to permit release of volatile hydrocarbons without deleterious gas-pocket formation. The type of furnace and optimum heating cycle (i.e., temperature and rate of change of temperature) will be determined.
- **Develop high-temperature heat treatment process.** The optimum heat treatment process is one that achieves the best trade-off between full graphitization of the compact and the allowable time at temperature that can be sustained by the fuel particles without damage. This task will determine the optimum parameters for high-temperature heat treatment of the compacts.
- **Coated-particle quality verification.** Perform QC inspections/tests to verify that coated-particle quality is not compromised in any way by the thermosetting-matrix compacting process.

Fabricate and inspect/test compacts for the multicell capsule shakedown test. Fuel particles from selected experimental coating runs will be fabricated into compacts for the shakedown capsule. Multiple compact lots will be necessary because of the different fuel variants to be included in the test. Thermosetting resin compacts will also be made from German-made fuel particles, if available, for inclusion in the shakedown capsule. The compacts will be inspected/tested in accordance with the requirements of the fuel product specification for the shakedown capsule.

Fabricate compacts for the fuel performance irradiation test. Fuel particles will be fabricated into compacts for the fuel performance irradiation test. Multiple compact lots will be necessary because of the different fuel variants to be included in the test particles. The compacts will be inspected/tested in accordance with the requirements of the fuel product specification for the fuel performance irradiation test.

Fabricate compacts for fuel qualification irradiation tests. Fuel particles will be fabricated into compacts for the fuel qualification irradiation tests. Only reference fuel particles will be included in the fuel qualification tests, so only a single compact lot will be necessary. The compacts will be inspected/tested in accordance with the requirements of the fuel product specification for the fuel performance irradiation test.

A.8 FUEL PRODUCT AND PROCESS DOCUMENTATION

Completion of the initial fuel fabrication development and irradiation tasks is expected to provide pertinent technical information that supports the selection and implementation of the fabrication process used by the VHTR fuel vendor and that can supplement the VHTR fuel vendor's own licensing/ qualification data in the topical report supporting VHTR licensing. Using this information, the program will finalize the top-tier fuel product specification. In addition to the fuel product specification, process modeling, test data, development results, and QC information will need to be thoroughly documented to define the processes that will successfully make and test the fuel that will consistently meet the product specifications.

Fuel product specification. Prepare a fuel product specification for the VHTR that incorporates the fuel properties and inspections/tests that have been determined from the fuel development effort to be essential to acceptable fuel performance during irradiation and accident conditions.

Fuel process development reports. Generate process development reports that document the work performed, modeling and analytical results, the final process parameters, and the allowable process variations (to the extent known) for each unit operation.

Quality control and testing development report. Generate a report that documents the QC methods and tests that were developed for enhanced characterization of gas reactor fuel. Itemize the QC data requirements determined necessary to demonstrate that the fuel meets the product specification requirements.

A.9 TECHNOLOGY FOR MASS PRODUCTION OF COATED-PARTICLE FUEL

Production QC methods. The QC methods for high-speed, nondestructive inspection of coated-particle fuel for key attributes will be developed. Wherever possible, methods that can be fully automated and integrated with the process line to provide real-time inspection and feedback of information to process controllers will be developed.

Production fuel fabrication. Use of high-speed, automated equipment with integrated process information feedback is thought to be the best way to meet NRC requirements for fuel process and product information and minimize the overall cost of the fuel. Waste minimization must be addressed in parallel with production planning to meet regulatory requirements and ensure that fuel processes do not need to be reengineered to eliminate specific chemicals or waste streams.

- **Develop conceptual design for an automated fuel production facility.** Develop automation and process documentation/feedback technologies for application to critical unit processes.
- **Develop waste minimization plan.** Evaluate all unit processes for potential waste reduction/minimization adjustments that will not adversely affect fuel quality.

Fuel production cost. Perform a cost study that accurately estimates the fuel fabrication cost for a commercially viable VHTR by using the reference process with automated equipment and QC methods.

A.10 SCHEDULE AND COST

Schedule basis. The basis for generation of the schedule and costs reflected in the dates below includes the following assumptions:

- The fuel manufacture effort start date is October 1, 2002.
- All necessary funding will be available and distributed to appropriate contractors on the start date.
- The development program schedule is not to be limited by funding constraints.
- The cost figures provided are not stand-alone estimates (i.e., coating depends on availability of kernels, coating scale-up requires model and parametric test data, and compacting requires coated particles).

The irradiation test plan for fuel qualification includes a capsule shakedown test, a fuel performance test, and a series of three fuel qualification tests. The schedule for these tests is shown in Sect. 3.2 of this program plan. The QC methods development must be completed early to support coating and compacting development.

Major fuel manufacturing milestones.

Task deliverable	End date
1. Initial coating process model	January 2003
2. First 10 kg of LEU UCO kernels and 10 kg of NUCO kernels	June 2003
3. Additional 40 kg of LEU UCO kernels	October 2003
4. Additional 30 kg of NUCO kernels	March 2004
5. Test specification and fuel product specification for capsule shakedown test and fuel performance test	June 2003
6. Complete small-coater tests/particles for capsule shakedown test	November 2003

Task deliverable	End date
7. QC methods development	December 2003
8. Compacts for capsule shakedown test	April 2004
9. Complete setup of large coater	September 2004
10. Fuel particles for fuel performance irradiation test	January 2005
11. Coating process model validation	January 2005
12. Compacts for fuel performance irradiation test	July 2005
13. Test specification and fuel product specification for fuel qualification tests	January 2007
14. Fuel particles for fuel qualification tests	February 2008
15. Fuel compacts for fuel qualification tests	May 2008
16. Fuel process and QC methods development reports	March 2007
17. Fuel product specification for GT-MHR fuel production	May 2008
18. QC methods automation report	September 2006
19. Process automation conceptual design	May 2006
20. Fuel production waste minimization plan	July 2006
21. Production fuel cost evaluation	November 2006

Cost breakdown by fiscal year.

Work category	Cost [($K) including NMCA (2%) and QA (6%)]						
	FY-03	FY-04	FY-05	FY-06	FY-07	FY-08	Total
Kernel manufacture	1021	1841	80	24	0	0	2966
Coating development	1348	1565	392	0	0	130	3434
Compacting development	670	221	86	0	0	76	1053
QC development	3093	244	0	0	0	0	3337
Irradiation test and fuel product specifications	432	43	36	41	102	43	697
Product/process documentation	0	81	24	44	116	86	351
Mass production analysis	0	2870	2859	2997	77	0	8802
Subtotal	6563	6866	3477	3106	294	335	20640

Appendix B. DETAILED FUEL AND MATERIALS IRRADIATION TASK LISTING

The high-level fuel and materials irradiation tasks discussed in Sect. 3.2 comprise a set of irradiation capsules. In the course of defining the irradiation activities, options in both the High-Flux Isotope Reactor (HFIR) and the Advanced Test Reactor (ATR) were defined in terms of technical characteristics, cost, and schedule. An Idaho National Environmental and Engineering Laboratory (INEEL) concept for a six-cell capsule to be operated in one of the large B holes of ATR was selected as the reference for the plan. In the discussions leading to the selection of a reference set of capsule irradiations to be conducted, it was agreed that, with one exception, this concept would be used for all of the capsules. That exception is noted and discussed in Sect. B.1, describing the planned capsule irradiation locations. The reference location and capsule concept were selected on the basis of limited design information and will require further review before a final commitment. The sequence of capsule irradiations identified for the plan is defined in Sect. B.2. The capsules are identified by task numbers consistent with the numbers used in Sect. 3 of this report. Because the activities required to define and conduct an irradiation are common to all the capsules, a generic detailed sequence of tasks was developed that is applicable to all the irradiations. This detailed list is provided in Sect. B.3. A number of topics were identified and addressed in the course of developing the irradiation, as discussed in Sect. B.4.

A set of goals, assumptions, and requirements for fuel and materials irradiation, consistent with those for the overall program provided in Sect. 2, was produced to guide development of the irradiation plan, as listed below.

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 postirradiation examination (PIE) and safety testing.

Assumptions

- Accelerated irradiation (up to 3X 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.
- Material sample irradiations can be conducted in conjunction with fuel irradiation without requiring additional capsules.
- Fuel fabrication capability is established to provide fuel samples representative of high-volume production for qualification testing.

Requirements

- Establish the range of irradiation conditions (i.e., 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 reactor rigs for irradiation testing of coated-particle fuel

*Gaseous constituents and impurity levels.

- Establish and conduct a fuel and materials irradiation activity that will
  - — provide for independently controlled and monitored cells within an irradiation capsule
  - — provide control capability to maintain conditions within the allowed tolerances
  - — provide for online monitoring of release of indicator fission product gases (e.g., krypton and xenon isotopes)
  - — provide a test design that will allow postirradiation measurement of integrated metallic fission product release (e.g., silver and cesium) from each cell during irradiation
  - — provide sufficient data to qualify the fuel for normal operation over the required range of irradiation conditions and support code and model development and validation
  - — provide irradiated fuel and material specimens as required to support PIE, postirradiation phenomenological testing, and safety testing activities

B.1 IRRADIATION LOCATIONS

As noted, the selected reference capsule to be used in all but one of the irradiations would be placed in one of the large B holes of the ATR. This capsule will contain six gas-tight cells that can be independently controlled for temperature and separately monitored for fission product gas release, with each cell capable of containing six fuel compacts. The exception, identified as AGR-4 in the following subsection, would utilize the single-cell capsule design previously used for irradiation of the NPR-1A capsule and would be placed in one of the small B holes.

The B holes in ATR are located in four triangular arrays, with each array comprising two small B holes and one large B hole. The "B" hole arrow in Fig. B.1 points to one of the eight small B holes, which are adjacent to the driver fuel. The four large B holes, located further from the driver fuel in the beryllium reflector, have a higher ratio of thermal to fast flux. Preliminary reactor physics calculations conducted by INEEL for the large B holes show a ratio of burnup to fast fluence that is very well matched to gas-turbine modular helium reactor (GT-MHR) conditions [maximum burnup 27% fissions per initial metal atom (FIMA) and maximum fast fluence 6×10^{25} neutrons/m^2] at the end of the irradiation. The physics calculations will be refined when the actual fuel loadings are known for each capsule, but the preliminary results with regard to the close match to the GT-MHR spectrum would not be expected to change significantly.

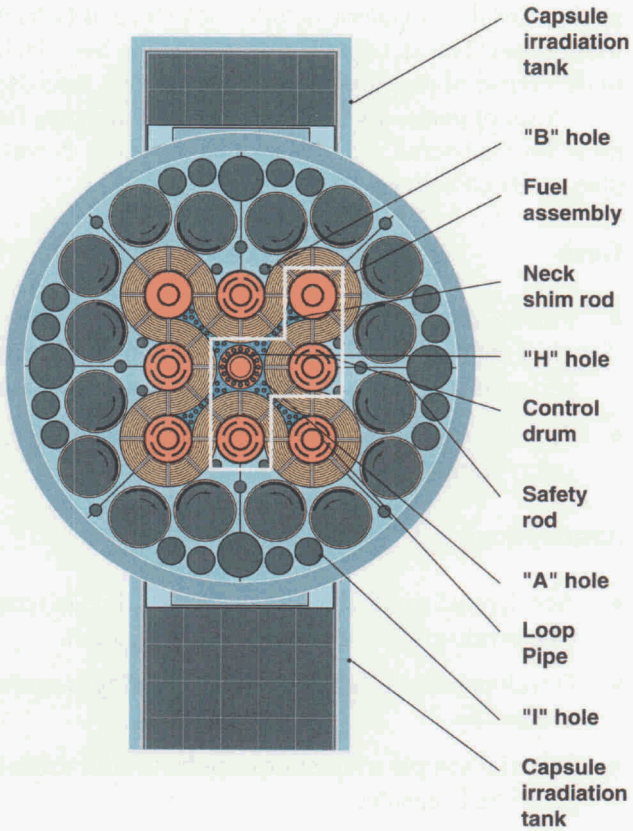

Fig. B.1. ATR cross section.

B.2 PLANNED IRRADIATION CAPSULES

The capsule irradiations listed in the following table were identified based on interactions of the fuel and materials irradiation group with the other working groups in the course of developing this plan. One

of the key issues in establishing this list was the advisability of conducting a shakedown irradiation of the capsule because it is a new, relatively complex design. As indicated in Table B.1, a shakedown capsule was adopted for the plan. The considerations with regard to inclusion of this capsule are addressed in Sect. B.4.

Table B.1. Planned irradiation capsules

Capsule	Task description[a]	Objective/expected benefits
AGR-1	3.2.1 Shakedown/Early Fuel—Contents to include compacts made from early small-coater particles, possible compacts made from German particles, as well as possible unbonded particles and material samples	Gain experience with multicell capsule design, fabrication, and operation and reduce chances of capsule or cell failures in subsequent capsules,[b] early data on irradiated fuel performance,[c] support development of a fundamental understanding of the relationship between fuel fabrication process and fuel product properties and irradiation performance
AGR-2	3.2.2 Performance Test Fuel—Contents to include compacts containing particles made in large coater from key variants in coated particles (e.g., IPyC permeability, anisotropy, uranium dispersion in buffer, continuous vs interrupted coating), possibly fuel performance modeling material samples, common cell temperatures in four central cells	Provide irradiated fuel performance data and irradiated fuel samples for safety testing and PIE for key fuel product/process variants to broaden options and increase prospects for meeting fuel performance requirements and to support development of a fundamental understanding of the relationship between fuel fabrication process and fuel product properties and irradiation performance
AGR-3	3.2.3 Fission Product Transport—Contents to include compacts of LEU UCO and NUCO particles seeded with designed-to-fail (DTF) fuel (LEU UCO and NUCO separately), unbonded kernels, DTF particles	Provide irradiated fuel performance data and irradiated fuel samples for safety testing and PIE. Data on fission product gas release from failed particles, fission product metal diffusion in kernels, and gas and metal diffusion in coatings for use in development of fission product transport models
AGR-4	3.2.4 Fission Product Transport—Contents to include DTF driver fuel and specialized samples of compact matrix and graphites	Provide data on fission product diffusivities and sorptivities in compact matrix and graphite materials for use in development of fission product transport models
AGR-5	3.2.5 Fuel Qualification—Contents to include a single fuel type made using process conditions and product parameters considered to provide best prospects for successful performance based on process development results and available data[d] from AGR-1 and AGR-2, variations in cell irradiation temperatures per test specification	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

Table B.1. (continued)

Capsule	Task description[a]	Objective/expected benefits
AGR-6	3.2.6 Fuel Qualification—Contents to include same fuel type as used in AGR-5, variations in cell irradiation temperatures per test specification	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
AGR-7	3.2.7 Fuel Performance Model Validation—Contents to include same fuel type as used in AGR-5. The irradiation would cycle the fuel thermally and be designed so that some measurable level of fuel failure would occur	Provide irradiation fuel performance data and irradiated fuel samples for safety testing 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 and -6 in support of plant design and licensing
AGR-8	3.2.8 Fission Product Transport 3—Contents to include compacts seeded with LEU UCO and NUCO particles with missing buffers, unbonded reference particles, different temperatures among cells, and to include temperature cycling	Provide irradiated fuel performance data and irradiated fuel samples for safety testing and PIE to determine material properties and fission product gas and metal releases from compacts with known quantities of failed particles for use in validation of fuel performance modeling and fission product transport codes

[a]All planned irradiation capsules except AGR-4 contain six independently controlled and monitored cells.

[b]If substantial early capsule failures (e.g., temperature control, gas monitoring, and instrumentation) are observed, the capsule could be removed and examined to determine required modifications in subsequent capsules.

[c]Particle variants tested should parallel those of subsequent large-coater product irradiations to the extent possible, potentially providing key data on metallic release, safety testing performance by end of first large-coater product irradiation (performance test fuel) in support of proceeding with fabrication of qualification test fuel as soon as possible.

[d]The decision to proceed with fabrication of qualification test fuel will be taken as early as possible based on information available at the time, which may include full irradiation of AGR-1 plus some heat up and fission product metal release data on AGR-1 fuel, as well as in-pile gas release data from AGR-2.

B.3 DETAILED IRRADIATION TASKS

It was necessary to produce a detailed sequence of the required tasks to develop an accurate estimate of the time and cost to conduct the capsule irradiations. Because the sequence, duration, and cost of these tasks are not expected to vary significantly for irradiations using the same capsule design and location, a generic set was produced and used in the plan for each of the capsules identified in Sect. B.2. Capsule AGR-4 was adjusted to reflect the cost and schedule differences (design, fabrication, and irradiation) associated with a single-cell capsule in the small B hole. In the course of developing the detailed tasks and identifying a reference capsule and location, data were developed for both ATR and HFIR. The ATR data are based on a six-cell capsule in the large B hole, as described earlier, and the HFIR data were for a single-cell capsule of the design used for the HRB-21 irradiation conducted previously. The ATR data in Table B.2 were used in the plan. For a given capsule, the "X" in the detailed task listing would be replaced by the corresponding number for the capsule; for example, irradiation of capsule AGR-2 would be Task 3.2.2.7.

Table B.2. Detailed capsule irradiation tasks and deliverables

Task title/description	Duration (months)		Cost ($K)		Deliverables	Interfaces
	ATR	HFIR	ATR	HFIR		
3.2.X.1 Test specification—Led by an organization representing the views of the fuel fabricator(s) and reactor designer(s), with review and concurrence by other affected program participants, this task will specify the test articles and the conditions and results needed to support fuel fabrication and model development and the plant design and licensing.	2	2	50	50	Test specification document—The document will include definition of test articles to be included in the capsule, required operating conditions (including tolerances), and required data (including accuracies) to be produced by the test.[a] *(Final version distributed 1 month after completion of task.)*	Input from fuel fabrication organizations, GT-MHR and/or PBMR designers, fuel and fission product transport modeling/code developers, and organizations conducting PIE or safety testing, as appropriate for a specific irradiation, will be necessary to assure that the requirements for the irradiation address the needs of the program
3.2.X.2 Capsule and supporting systems design, operational and functional requirements—Conducted by the organization performing the irradiation, with review and concurrence by other affected program participants, this task will establish the detailed requirements necessary to proceed with capsule and supporting systems design in accordance with the test specification.	2[b]	2[c]	100	40	Irradiation capsule and supporting systems design, operational and functional requirements document—The document will include general design requirements associated with the service conditions of the capsule in the reactor, design and functional requirements specific to the capsule and its supporting systems, and provisions for QA.[d] *(Final version distributed 1 month after completion of task.)*	Review by fuel fabrication organizations, GT-MHR and/or PBMR designers, fuel and fission product transport modeling/code developers, and organizations conducting PIE or safety testing, as appropriate for a specific irradiation, will be necessary to assure that the capsule and supporting system requirements will meet the intent of the test specification

Table B.2. (continued)

Task title/description	Duration (months)		Cost ($K)		Deliverables	Interfaces
	ATR	HFIR	ATR	HFIR		
3.2.X.3 Capsule and supporting systems design (new capsule/proven capsule)—This task will establish the detailed design and procurement specifications necessary to proceed with capsule fabrication/assembly and establish the needed supporting systems for either a new capsule design or replication of a proven capsule design	4/2	8/4	1750/ 500[e]	500/ 250[f]	Irradiation capsule and supporting systems design and planned operation description document—This document will provide sufficient information on the as-built design and planned operation of the capsule and supporting systems to allow calculation of the capsule and test article operating conditions for pre- and postirradiation performance predictions.[g] *(Preliminary version distributed to affected program participants 1 month before completion of "capsule fabrication/assembly" task; final version distributed 1 month after completion of "review/approval of final design and fabrication data packages" task.)*	Review by fuel fabrication organizations, GT-MHR and/or PBMR designers, fuel and fission product transport modeling/code developers, and organizations conducting PIE or safety testing, as appropriate for a specific irradiation, will be necessary to assure that the capsule design and planned operation will meet the intent of the test specification
3.2.X.4 Capsule and supporting systems fabrication/assembly—This task will include procurement or fabrication of capsule components in accordance with the specifications, installation or refurbishment of supporting systems as necessary, and assembly of the capsule, including the test articles, ready for insertion into the reactor.	8	5		300		
3.2.X.5 Receipt of test articles—This event includes the receipt of all test articles (compacts, pebbles, loose particles, material samples) to be incorporated into the capsule, with the duration shown being the lead time required prior to completion of capsule assembly.	2	2	0	0	Test article receiving inspection report—This report will document the condition of test articles as received and confirm that they are in compliance with the descriptions provided in the test specification.	Test articles, along with necessary characterization data and QA documentation will be provided by organizations fabricating the fuel and material samples

Table B.2. (continued)

Task title/description	Duration (months)		Cost ($K)		Deliverables	Interfaces
	ATR	HFIR	ATR	HFIR		
3.2.X.6 Review/approval of final design and fabrication data packages[h]—This task includes review and concurrence by other affected program participants, conducted in parallel with capsule design and fabrication except for the duration shown	1	1	25	40	Documented concurrence of affected program participants that the as-built capsule is in compliance with the test specification. Irradiation capsule ready for installation into the reactor and supporting systems ready to begin operation for control and monitoring of the capsule.	Review by fuel fabrication organizations, GT-MHR and/or PBMR designers, fuel and fission product transport modeling/code developers, and organizations conducting PIE or safety testing, as appropriate for a specific irradiation, will be necessary to assure that the as-built capsule will meet the intent of the test specification

Table B.2. (continued)

Task title/description	Duration (months)		Cost ($K)		Deliverables	Interfaces
	ATR	HFIR	ATR	HFIR		
3.2.X.7 Irradiation—This task addresses all activities associated with irradiation of the capsule, 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 capsule in its storage location for cooldown.	24[i,j]	[k]	350/year[l]	[m]	External data acquisition—A system will be established to support secure near real-time access to irradiation data by authorized program participants in accordance with the requirements for reporting data during the irradiation. Capsule irradiation data report—This document will provide detailed time-dependent data on the irradiation conditions (estimated axial flux distributions at the capsule location, measured capsule temperature distributions, individual cell gas mixtures, etc.) and fuel performance (R/B for relevant isotopes for each cell). *(Interim versions produced based on data collected through the end of each 1 month interval of irradiation, to be distributed to affected program participants 2 weeks after end of each interval. Draft final version to be distributed for review 2 months after end of irradiation, final version distributed four months after end of irradiation.)*	Review of irradiation data by authorized program participants with feedback to the organization conducting the irradiation if and as appropriate. Review of draft irradiation data reports by fuel fabrication organizations, GT-MHR and/or PBMR designers, fuel and fission product transport modeling/code developers, and organizations conducting PIE or safety testing, as appropriate for a specific irradiation, will be necessary to assure that the reported data will meet the intent of the test specification
3.2.X.8 Cooldown—This task addresses storage of the capsule until the decay heat and radiation levels are sufficiently low to proceed with capsule disassembly and shipping.	3	3	0	0		

Table B.2. (continued)

Task title/description	Duration (months)		Cost ($K)		Deliverables	Interfaces
	ATR	HFIR	ATR	HFIR		
3.2.X.9 Disassembly and shipping—This task includes disassembly of the capsule to the extent necessary for shipment, and shipping the disassembled components to the organizations conducting PIE and safety testing.	2	2	100[n]	150[o]	Distribution of irradiated test articles—Test articles packaged and distributed as required for PIE and safety testing. *(articles shipped at end of task)* Capsule disassembly data report—This document will provide a description of procedures used and data obtained during disassembly of the capsule and preparation of test articles for shipment or transfer (visual observations of conditions of capsule internals, etc.). *(Draft version distributed for review 1 month after end of task, final version distributed 2 months after end of task)*	Receipt of irradiated test articles by organizations conducting PIE or safety testing will facilitate the start of those activities Review of draft capsule disassembly data reports by fuel fabrication organizations, GT-MHR and/or PBMR designers, fuel and fission product transport modeling/code developers, and organizations conducting PIE or safety testing, as appropriate for a specific irradiation, will be necessary to assure that the reported data will meet the intent of the test specification

[a]Input from fuel fabrication organizations, GT-MHR and/or PBMR designers, fuel and fission product transport modeling/code developers, and organizations conducting PIE or safety testing, as appropriate for a specific irradiation, will be used to communicate to the organization that will design the test apparatus in an irradiation test specification the information needed to design the test apparatus and how to conduct the irradiation. The information will include:

- Test objectives
- Test articles (compacts/pebbles, loose particles, material samples) description
- Test requirements
  - Basis for specified test conditions
  - Allowable ranges of test conditions (irradiation conditions and operational requirements)
  - Measurement requirements
  - Test results and acceptance limits
  - Safety and quality assurance (QA) requirements
  - Codes and standards
- Documentation requirements
- Reporting requirements

[b]Conducted in parallel with the test specification development except for the duration shown above.

[c]Particle variants tested should parallel those of subsequent large-coater product irradiations to the extent possible, potentially providing key data on metallic release, safety testing performance by end of first large-coater product irradiation (performance test fuel) in support of proceeding with fabrication of qualification test fuel as soon as possible.

[d]The testing organization will respond to the irradiation test specification with a capsule and supporting systems design, operational, and functional requirements document. This document will describe in a one-for-one manner how the testing organization will meet the test specification and how the capsule design will meet the test reactor requirements. It will contain (with reference to supporting documentation for details as appropriate):

- Description and capabilities of the test reactor facilities and equipment to be employed
- Capsule design requirements
- Supporting systems design requirements (e. g., R/B measurement system)
- Requirements for capsule instrumentation and control of the irradiation
  - Online instruments
  - Measurements accessible after capsule disassembly (e. g., flux wires)
- Requirements for reporting of irradiation data
  - During irradiation
  - After completion of irradiation
- Safety requirements (provisions for preventing excursions outside the test envelope and for mitigating the consequences of possible accidents)
- QA plans

[e]Both: Includes fabrication/assembly costs for an instrumented lead containing up to six independently gas swept cells and fission gas monitoring. Proven: Cost for exact duplicate of first capsule. The magnitude of any changes from the first design will dictate cost increases.

[f]The estimates are for a single cell (approximately 500-cm^3 volume) swept gas capsule.

[g]The irradiation capsule design and planned operation description document will describe how the capsule design and planned operation meets the requirements of the capsule design, operational, and functional requirements document and provide information necessary for analysis of test data by other program participants. It will contain (with reference to supporting documentation for details as appropriate):

- Capsule design description
- Supporting systems design description (e. g., R/B measurement system)
- Capsule instrumentation and control system description
  - Online instruments
  - Measurements accessible after capsule disassembly (e. g., flux wires)
- An analysis of the ability of the test hardware to meet the requirements
  - Temperatures, burnup, fast neutron fluxes and fluences, rates, and temperature gradients, as specified in the irradiation test specification
- Plan for capsule operation
- Plan for reporting of irradiation data
  - During irradiation
  - After completion of irradiation
- Plan for capsule disassembly and shipping
- Provisions for preventing excursions outside the test envelope and for mitigating the consequences of possible accidents
- QA documentation for design and QA requirements for operation and data collection

[h]Includes review and concurrence by other affected program participants, conducted in parallel with capsule design and fabrication except for the duration shown.

[i]Duration of the irradiation will depend on the desired burnup and fluence as well as allowed maximum acceleration factor.

[j]If the irradiation schedule spans the reflector replacement an additional 6 months would be needed.

[k]Duration of irradiation will depend on desired burnup, fluence, and acceleration factor. In an RB position (like HRB-21/22) fast fluence ($E > 0.1$ MeV) accumulates at $\sim 1.0 \times 10^{25}$ neutrons/m^2 per 24-d cycle.

[l]Includes neutron charges, analysts time, and online real-time data for 6 cells.

[m]Neutron costs for an R/B position for non-Office of Science (SC) programs are $90K/cycle. Surveillance costs should be approximately $40K/cycle.

[n]$50K for disassembly and $50K for shipping to ANL-W.

[o]Shipment of capsule from HFIR to Irradiated Fuels Examination Laboratory (IFEL) and disassembly of capsule to point of Test Article recovery.

B.4 CONSIDERATIONS REGARDING INCLUDING A SHAKEDOWN CAPSULE

The list of capsules provided in Sect. B.2 includes an initial shakedown capsule, identified as AGR-1. Because of the potential near-term impact on program funding and schedule, the inclusion of a shakedown capsule was given careful scrutiny. The considerations involved and the basis for inclusion of a shakedown capsule are provided below.

B.4.1 Recommendation

On the basis of the considerations discussed below, it is recommended that the advanced gas reactor (AGR) fuel development and qualification program plan include a shakedown capsule that is prototypical of the six cell capsules in the ATR "B hole" that are envisioned for the majority of subsequent irradiations. It is further recommended that the fuel used in the capsule include early small-coater product variants to be determined at the time to give the best chance for at least one successful result and compacts fabricated from German fuel particles using both thermoplastic matrix (tested in the MHR-1 irradiation in Petten) and thermosetting matrix. To minimize demand for scarce near-term program funding, the fuel characterization would be limited to that being performed for fuel process development plus the minimal additional characterization (if any) considered necessary to begin the irradiation. Additional characterization would be performed later if warranted by the situation at the time. The duration of shakedown capsule operation would depend on the situation, but it is recommended that the reference case assume continued operation to approach full burnup in the fuel.

B.4.2 Considerations Against Inclusion of a Shakedown Capsule

1. Competition for funding in the near term will delay other activities on the critical path to irradiation of qualification test fuel.
2. Current draft schedule will require ~$150K in FY 2003 to prepare test specifications and requirements documents.
   - Additional ~$2M in FY 2004 to complete design and fabrication of capsule and initiate irradiation
   - Additional ~$200K in FY 2004 to fabricate and characterize fuel to be loaded into the capsule
3. It would add significant later costs of ~$350K/year for irradiation, ~$500K for limited PIE and safety testing, and ~$150K for additional fuel characterization if data are to be used by the program.
4. It would contribute to a perception that the program is planning too many irradiations.

B.4.3 Considerations Favoring Inclusion of a Shakedown Capsule

1. The risk of failure in a complex untested new capsule design is significant and could lead to major program setbacks if failures are experienced in the performance test fuel capsule. Note that there are difficulties most of the time with the first capsule of a new design. Examples of problems encountered include leakage between cells; failure of sensors because of shorts and loose connections; inability to control temperature; chemical reactions between thermocouple sleeves and fuel; and in some cases, total failure of the capsule such that the test had to be terminated. A shakedown capsule in an early, less critical irradiation can reduce the chances of capsule failures with major impact on the program. If early major failures are experienced, the capsule can be removed and disassembled to identify the cause and modify the design prior to the fabrication of the performance test capsule. Minor failures may provide sufficient information to guide subsequent capsule modifications, while the shakedown capsule continues operation based on overall cost/value considerations.

2. A shakedown capsule would need to include fuel similar to the fuel to be tested in subsequent capsules, so that the temperature and heat flux distributions are sufficiently prototypical to represent a valid test of the design; thus, the incremental cost of including early small-coater fuel variants in a shakedown capsule will be relatively small. If early fuel samples are used in the capsule, the duration of the irradiation can be adjusted to maximize the value/cost ratio for the capsule. Inclusion of early fuel can substantially increase the value of conducting the shakedown test in the following ways (for the purposes of this discussion it is assumed that the shakedown capsule precedes the performance test capsule by 9 to 12 months and the fuel product variants are similar to those chosen for the performance test):
   - Early in-pile release data may be used to enhance the selection of the fuel variants to be used in the fuel performance test, increasing the prospects of success of that capsule
   - If the shakedown capsule conducts an extended irradiation of the fuel (e.g., 18–24 months), there would still be time to conduct some metallic release and heatup testing on the best performing fuel prior to completion of the performance test capsule irradiation. This would provide a stronger basis (reduced risk) for proceeding with the fabrication of the qualification test fuel at the completion of the performance test irradiation based on in-pile gas release data for large-coater fuel supported by metallic release and heatup data for similar small-coater fuel.
   - If substantial delays are encountered in producing fuel with a large coater, good quality positive data available from the shakedown capsule could take on added importance. Furthermore, in the event of such a delay, if the product from the large coater, as determined by preirradiation characterization, is sufficiently similar to the fuel tested in the shakedown capsule, it may be justifiable to omit the performance test of large-coater fuel and to proceed directly to qualification testing of the large-coater fuel.
   - Additional irradiation data from small-coater fuels will be of value in terms of strengthening confidence in the understanding of fuel performance.
3. Assuming the irradiation performance in one or more of the cells is positive, the test would produce early tangible results that could enhance confidence in the program and improve early prospects for industry support and increased funding.

B.4.4 Incremental Cost of a Shakedown Capsule

The following estimate of incremental cost is based on the assumption of a six-cell capsule containing early small-coater fuel loaded into the large B hole in ATR. The total incremental cost ranges from ~$850K for a short irradiation that identifies major capsule design problems to ~$2.1M for a full irradiation that would provide valuable fuel performance information. Also, as noted below, the net first-of-a-kind capsule design and fabrication cost of $1250K would be moved forward 9–12 months.

1. Test Fuel Fabrication—It is assumed that the compacts included in the capsule would be made from coated particles fabricated during the coating development work in the small coater and from archived German fuel particles. There would not be a full characterization of the fuel to be loaded into the capsule, only the characterization that would be done for coating and compacting process development. If the results of the irradiation were seen to be of sufficient technical and programmatic value, additional characterization of archived product would be performed later. On this basis, the incremental cost for fabrication of the shakedown test fuel compacts would be ~$200K. The cost of the additional characterization would be ~$150K.
2. Capsule Design/Fabrication Costs—Because this would be the first capsule of this design, the costs would include $150K for the test specification and design requirements and $1750K for design and fabrication, for a total cost of $1.9M. Because the first-of-a-kind cost for the capsule would have to be paid in any case, the net incremental cost of the capsule would be $150K plus $500K for a repeat capsule, totaling $650K.

3. Irradiation Costs—The irradiation costs are estimated at $350K/year, thus ranging from a low of ~$100K for an early removal of the capsule to ~$700K for a full irradiation.
4. Other Costs—Disassembly and shipping is estimated at $100K. If the capsule approached full burnup, it may be desirable to conduct some limited PIE and safety testing [e.g., leach-burn-leach ($32K per compact) and limited heatup testing ($250K/compact)].

B.5 COMPARISON OF IRRADIATION FACILITIES FOR COATED-PARTICLE FUEL TESTING

This section provides a brief overview of irradiation facilities that could be used to support U.S. coated-particle fuel testing under the AGR Fuel Development and Qualification Program. Because each facility has experience in gas reactor fuels irradiations, each is reviewed with emphasis on the physical sizes of the potential capsules and the availability of the reactor to perform such irradiations in the near term than on the ability of the reactor to produce the design service conditions relevant to the GT-MHR or PBMR (Table B.3).

BR-2. The BR-2 reactor is a materials test reactor in Mol, Belgium.[1] It uses highly enriched uranium (HEU) UAL_x/Al cermet fuel plates to produce very high fast [3.5×10^{14} neutrons/cm^2-s (E > 1 MeV)] and thermal neutron fluxes (10^{12} neutrons/cm^2-s), which makes it a good candidate for gas reactor fuels testing. The facilities have irradiation test rigs (~15 mm ID and 400 mm long) that can be used to irradiate coated-particle gas reactor fuel forms. They have adequate flux, fluence, and temperature characterization for the capsule and have the infrastructure needed for capsule disassembly and PIE. The capsule size precludes them from irradiating pebbles, but they could handle about six to eight General Atomics (GA) fuel compacts.

IVV-2M. The IVV-2M is a 15-MW water-cooled reactor that has been used in Russia for a variety of coated-particle testing.[2] Four different test rigs have been used to test specimens ranging from particles to compacts to spheres. The CP ampoule is a noninstrumented rig that can hold 10 to 13 graphite disks (15 mm in diameter and 2 mm thick), each of which can hold 50 particles. The rig can also hold coated particles in axial holes, 1.2 mm in diameter, and a uniform volume of coated particles, 12 to 18 mm in diameter and 20 to 255 m high, in a graphite matrix. Another rig, termed a CP hole is 27 mm in diameter and can handle six to eight capsules. A third rig, identified as ASU-8, is a 60-mm hole that can handle three compacts. The largest channel available is Vostok, which is 120 mm in diameter and contains four cells. All of these rigs can irradiate fuel at representative temperatures, burnups, and fluences for PBMR and GT-MHR. PIE facilities are also available. There is a large degree of flexibility in the testing options at IVV-2M. Their rigs can handle particles, compacts, and spheres. They are currently irradiating spheres for China and have plans in the future for PBMR spheres; however, restrictions by the U.S. Department of State currently prevent use of IVV-2M under U.S.-sponsored programs.

HFR Petten. The High Flux Reactor (HFR) in Petten, the Netherlands, is a multipurpose research reactor that has many irradiation locations for materials testing.[3] It has been the workhorse for irradiation of spheres for the German PBMR project in the 1970–1995 time frame. It has also irradiated GA compacts for the U.S. program in the late 1980s. They have two different types of irradiation rigs/locations in the facility: one that can accommodate compacts and one that can accommodate spheres. The REFA and BEST rigs are multicell capsules, 63 to 72 mm in diameter, that can handle four to five spheres in up to four separate cells. The TRIO or QUATTRO rigs/locations are ~32 mm in diameter and 600 mm in useful length. They can handle three or four parallel channels of compacts. For the three-channel configuration, about 30 compacts could in principle be irradiated in the rig. These rigs are currently dedicated to the EU-1 (sphere) and EU-2 (compact) irradiations under the HTR-F program in Europe. The current configurations of EU-1 and EU-2 are limited in the number of individually swept cells that are being used. In EU-2, only two cells are planned, one for German spheres and one for Chinese spheres. In EU-2, only one swept cell is planned for the U.S. compacts. In addition, there is a large axial flux gradient

Table B.3. Comparison of irradiation facilities for coated-particle fuel testing

Reactor	Location	Gas reactor experience	Flux, fluence temperature characterization, and fission gas release	Capsule space description	Infrastructure (PIE, SEM)	Comments
BR-2	Mol, Belgium SCK·CEN	Yes	Adequate	15-mm ID and 400 mm long	Yes	Too small for pebbles; can do six to eight compacts in a single capsule
IVV-2M	Russia	Yes	Noninstrumented	CP ampoule 10 to 13 disks (15-mm diam and 2 mm thick) that can each hold 50 coated particles	Yes	Large degree of flexibility; can handle particles, compacts and spheres. Currently busy with HTR-10 and then PBMR irradiations U.S. Dept. of State restrictions currently prevent its use in U.S.-sponsored programs.
			Yes	CP Hole, 27-mm diam, 6 to 8 capsules		
			Yes	ASU-8, 60-mm hole can handle three compacts		
			Yes	Vostok, 120-mm hole, four cells available		
HFR Petten	Petten Netherlands	Yes	Yes	REFA and BEST, 62- to 72-mm diam up to four cells, four to five spheres	Yes	Used for EU-1 irradiation in near term
				TRIO and QUATTRO, 31.5-mm ID, 600-mm useful length. Three or four parallel channels that could handle 30 compacts		Dedicated to EU-2 irradiation in near term. Only utilizing one of three positions available in each graphite body. Single swept capsule. Large flux and burnup gradient (40% spread max to min) over the useable length

Table B.3. (continued)

Reactor	Location	Gas reactor experience	Flux, fluence temperature characterization, and fission gas release	Capsule space description	Infrastructure (PIE, SEM)	Comments
HFIR	ORNL, USA	Yes	Yes	38-mm ID, 500 mm long	Yes	Too small for pebbles. Can handle 24 compacts. Single swept capsule. Large axial flux gradient
				24-mm ID, 500 mm long		Too small for pebbles. Can handle 16 compacts. Single swept capsule. Large axial flux gradient
ATR	INEEL, USA	Yes	Yes	89-mm ID (medium I hole), and 127 mm ID (NE Flux Trap), 760 mm long	Yes	Can handle six individually swept cells, two pebbles per capsule. Very flat burnup, fluence profile axially.
				38-mm ID (large B hole) 760-mm length		Can handle five individually swept cells, two graphite bodies per cell, each containing three compacts. Very flat burnup, fluence profile axially.
SAFARI	Pelindaba, South Africa	Yes	Yes		Yes	Used to conduct pebble irradiations for PBMR

across the useable length (40% spread maximum to minimum) that must be considered in the design of any experiment.

HFIR. The High Flux Isotope Reactor at Oak Ridge National Laboratory (ORNL) is a light-water cooled, beryllium-reflected reactor that uses HEU U-Al fuel to produce high neutron fluxes for materials testing and isotope production. It has been used extensively in the U.S. gas reactor program to irradiate coated-particle fuel. Two specific materials irradiation facilities are of note here. The large RB positions (of which there are 8) are 46 mm in diameter and 500 mm long and can accommodate capsules holding up to 24 compacts, (3 in each graphite body, 8 bodies axially) in a single swept cell. This configuration was used for the HRB-21 experiment, the last irradiation in the U.S. commercial program in the 1990s. The small VXF positions (of which there are 16) are 40 mm in diameter and 500 mm long. They can accommodate capsules holding up to 16 compacts (8 in each graphite body, 2 bodies axially) in a single swept cell. This configuration was used for the NPR-1 and NPR-2 irradiations, the last two irradiations at ORNL under the NP-MHTGR program in the 1990s. Capsules can be irradiated in the lower flux small VXF positions and then moved to the higher flux RB positions, as was done with NPR-1 and NPR-2. Neither of these positions can accommodate pebbles. A third facility, the large VXF positions (of which there are 6), are farther out in the reflector (and therefore have lower fluxes), but are 72 mm in diameter and also 500 mm long. As with the HFR, there is a large axial flux gradient that must be considered in the design of any experiment in any of these facilities.

ATR. The Advanced Test Reactor at INEEL is a light-water-cooled, beryllium-reflected reactor that uses HEU U-Al fuel in a four-leaf clover configuration to produce high neutron fluxes for materials testing and isotope production. The clover leaf configuration results in nine very high flux positions, termed flux traps. In addition, numerous other holes of varying size are available for testing. Of interest here are several holes that can be used to irradiate coated-particle fuel. The 89-mm-diam medium I-hole (of which there are 16) and the 100- to 125-mm-diam flux traps can accommodate pebbles. Specifically, the use of a medium I-hole early in the irradiation (if required because of the enrichment of the fuel), followed by transfer of the test train to the northeast flux trap, can provide irradiation conditions representative of the PBMR. Approximately 10 to 12 pebbles in 5 or 6 individually swept cells can be envisioned in the test train. The large B holes in ATR (of which there are four) are 38 mm in diameter` and 760 mm in length. They can accommodate five individually swept cells, with two graphite bodies per cell, containing up to three compacts per body. Thus, a total of 30 U.S. compacts can be irradiated in this location. Of special note here is the very flat burnup and fluence profile available axially in the ATR over the 760-mm length. This allows for nearly identical irradiation of large quantities of fuel. The ATR was used extensively during the NP-MHTGR program to irradiate targets (ATR-1, ATR-2, ATR-3, and ATR-4 series of experiments) and fuel (NPR-1A irradiation) in the early 1990s.

SAFARI. The SAFARI Reactor in Pelindaba, Republic of South Africa, is an isotope production and research reactor.[4] The core lattice is an 8×9 array, consisting of 28 fuel assemblies (each containing 19 flat U/Al alloy fuel plates), 6 control rods and a number of aluminum and beryllium reflector assemblies. The reactor is cooled and moderated by light water and operates at a maximum power level of 20 MW. In-core irradiation positions include six high-flux isotope production positions, two hydraulic, two pneumatic, and two fast transfer systems that are accessible during operation. Several other irradiation positions can also be accessed when the reactor is shut down. A large poolside facility allows for a variety of radiation applications. An intermediate storage pool and transfer canal allow for easy and safe transport of activated materials to a hot cell. Six neutron beam tubes are available, one of which is permanently used for neutron diffraction studies and another for neutron radiographic investigations. The SAFARI reactor is planned to be used in conjunction with the IVV-2M reactor for the qualification of fuel for PBMR.

B.6 GENERAL TOPICS ASSOCIATED WITH PLAN DEVELOPMENT

The following material was produced in the course of developing the reference irradiation location and sequence of irradiation capsules and is provided as supporting background material.

1. Types of Fuel Test Articles—Because of the need for fabrication process development in the case of the GT-MHR fuel (as opposed to replication of the last generation German fuel for PBMR), it is anticipated that at least the early irradiation testing would be for GT-MHR fuel. The smaller diameters of the GT-MHR fuel compacts relative to the PBMR pebbles allows more and better options for irradiations in ATR and also allows options for irradiation in HFIR. For these reasons, the schedule options in the table have been developed based on compact irradiations. Both the ATR and HFIR capsules could accommodate assemblies of unbonded particles as well, which may be desirable for early irradiations in support of process development and fission product transport modeling. The details for unbonded particle irradiations would be addressed in the course of design of irradiation capsule internals. HFIR would be unable to accommodate PBMR pebble fuel elements because their diameter is too large for the irradiation locations. ATR has locations large enough for pebble irradiations, but the flux spectrum is less favorable than for locations that can be used for compact irradiations. Considerable information regarding pebble irradiations in ATR was developed in support of the Exelon PBMR preapplication licensing interactions and can be accessed if pebble irradiations are needed in the future.

2. Irradiation Options for Early Feedback to Fuel Process Development—The information presented in Table B.2 is based on well-controlled and instrumented capsules with capability for irradiation of relatively large quantities of fuel and, in the case of the ATR capsule, multiple independently controlled and monitored cells. A question was raised regarding near-term options that could be conducted more quickly and cheaply to provide early feedback on irradiation performance of fuel and material samples. The general consensus of the working group is that the program would be better served by utilizing the kind of irradiation capability provided by capsules as envisioned in the table, recognizing that a near-term irradiation directed toward early feedback to fuel process development may have a greater variety of test articles that would need to be addressed in the detailed design of capsule internals, in comparison to later irradiations for demonstration and qualification testing.

   Based on initial results of the Fuel Manufacture Working Group, lead times for a well-controlled and instrumented capsule, as identified in the table, are expected to be well within the lead times required to produce early unbonded particles and/or compacts in the fuel process development activity. Shortening the irradiation time can be done for the capsules envisioned in the table, but it raises questions of effects of acceleration on irradiation performance. One approach for getting early feedback as well as data regarding acceleration effects would be to conduct parallel irradiations with partial duplication of test articles, having maximum acceleration in HFIR and closer to real time in ATR. In addition to providing early feedback, the results could establish a basis for acceleration of later irradiation of fuels for demonstration and qualification testing.

   Specific comments from ORNL are provided below:

   No "quick and cheap" way to do a fully instrumented fuel test in HFIR has been identified. If the program wants to irradiate small amounts of nonfuel materials (like SiC or graphite), it may be possible to use rabbit capsules in the target region of HFIR, where it is possible to obtain ~2.0×10^{25} neutrons/m^2 ($E > 0.1$ MeV) in one 24-d cycle. These could be done for tens to hundreds of thousands of dollars, depending on requirements. If that rate of fluence accumulation is too fast, something similar could probably be done out in an R/B position or even farther out in the reflector. It may also be possible to irradiate a few loose particles in rabbit capsules out in the reflector, but because these

would be uninstrumented, the temperatures would have to be calculated. Passive thermometry (like SiC monitors) could be used. The bottom line is that we cannot do "quick and cheap" fully instrumented capsules, but it may be possible to do some "quick and cheap" uninstrumented capsules with small quantities of materials.

3. Facility Capabilities and Constraints for Conducting Irradiations—Both ATR and HFIR have high-quality documentation of the characteristics and capabilities of the facilities that can be incorporated into the plan by reference. Summary discussions of the capabilities relevant to the AGR program are provided below.

   HFIR has the capability to run up to eight instrumented R/B capsules at a time, but some instrumentation additions/modifications would be required. When the NPR-1 and NPR-2 capsules were being irradiated, a total of four or five capsules were being irradiated in parallel (the others were for the Fusion Materials Program).

   In principle, the ATR has the capability to perform multiple irradiations in parallel. For compacts, the four large B-holes could be used for irradiations. The availability of a large B-hole would depend on when such an irradiation is scheduled and what other customers might be occupying or wanting to occupy the position. For pebbles, because of their size, the options are somewhat more limited. The Northeast and South flux traps can permit irradiation of pebbles to high-burnup design service conditions. The medium I holes and flux-enhanced large I holes have somewhat lower neutron fluxes and fluences and thus offer the potential as a position to start a high-burnup irradiation while keeping power generation in the fuel body to an acceptable level or with an increase in the respective lobe power the potential to achieve moderate levels of burnup/fluence. (Such an increase in lobe power is in principal an option if the programmatic importance of the work is high enough based on discussion with the ATR sponsor) . There is adequate capability in the automated thermal control system to allow two irradiations in parallel; however an additional gamma spectrometer system would be needed, which would not be a serious cost driver. The capability to display experiment data online, in nearly real-time currently exists here, as described in a color brochure containing sample Web pages. This feature has been factored into the ATR irradiation cost assuming six data streams (one stream for each cell).

4. Conceptual Design of Irradiation Tests and Instrumentation Needs—Irradiation testing generally consists of a test train that contains a number of fuel compacts or pebbles. The number of fuel specimens is a function of the reactor, the exact location of the irradiation, and the particular goals of the irradiation. Given the statistical nature of the fuel, a large number of fuel specimens are needed to demonstrate compliance with the fuel failure specification. This can be accomplished either by a number of smaller irradiations or a few larger irradiations, depending on the reactor to be used.

   A highly instrumented test train is used for the irradiation. Historically, both thermocouples and passive temperature monitors have been used to measure temperatures. Thermocouples are recommended because of their ability to track temperature in a continuous fashion. The thermocouples are not attached to the fuel specimens because of concern that the thermocouple material could be a source of contamination that might threaten the SiC layer. Instead the thermocouples are attached to graphite sleeves or holders, and detailed thermal calculations are performed to establish the fuel temperature. Such detailed calculations have been verified in previous irradiations. Flux wires and neutron fluence monitors are installed in the test train to measure the thermal and fast neutron fluence in the experiment. In the ATR, continuous-length flux monitors are installed on the periphery of the test train and removed every cycle to provide accurate flux and power data as input to detailed neutronics and thermal calculations that are performed to tract and monitor the progress of the irradiation.

Thermal control during the irradiation is accomplished using a gas gap between the graphite sleeve housing the fuel specimens and the outer metal cylinder that provides containment of the capsule from the reactor. Very high temperatures can be achieved using this approach. During the irradiation, a low flow of inert sweep gas is used to provide the correct thermal conductance to allow the fuel to be irradiated at the proper temperature. Usually this sweep/thermal control gas is helium. Small amounts of neon are used to change the overall conductance to compensate for uranium depletion while maintaining the fuel at the required temperature. With proper thermal design of the experiment, a temperature window of 400°C to 500°C around the nominal irradiation temperature can easily be established.

An important objective of the irradiation is to be able to measure the fission gas release from the fuel and be able to correlate it to the operating parameters in the irradiation. In the past, the fuel specimens have been "sniffed" for fission gas either collectively or individually, depending on the requirements. In principle either technique can be accommodated in the design of the test train. The sniffing gas is also used to transport any fission gases released from the fuel to a location outside of the reactor. There, an ion chamber with enough sensitivity to provide an indication of a single fuel particle failure line by a spike in its signal measures gross radiation in the line. In some cases, the isotopic content of the gas in the line is monitored online using a state-of-the-art fission product monitoring system. This system consists of a gamma spectrometer to provide a continuous measurement of the concentration of the various fission gas isotopes in the sweep gas. This approach was used quite successfully in the NPR-1A fuel irradiation conducted in 1991–1992 in the ATR. In other cases, the gas is collected in a sample vessel and measured offline. This approach has been historically used in Petten and in HFIR at ORNL. These instruments provide a complete time history of gas release. Such temporal information provides information on the source of the fission gas. Gas release early in the irradiation (i.e., from the start of the irradiation) is attributed to initially failed particles or contamination outside of the SiC layer. Release later during the irradiation is indicative of in-situ particle failure. The timing of the failure data can then be correlated to temperature, burnup, and/or fluence that, when coupled with PIE, can be used to determine the mechanisms responsible for the fuel failure. The R/B is the classical way to express the release of short-lived fission gases during irradiation. The online gamma spectroscopy system can provide R/B data on about six to eight of the short-lived noble gases. In ATR the offline grab sample generally can only provide R/B data for ^{85m}Kr because of radioactive decay of the other short-lived fission gases that occur from the time of sample collection to the time of analysis. Online systems are recommended because of the greater time resolution available with the spectrometer, and comparison of the results of more than one noble gas isotope available with online systems can be used to identify any isotope-specific measurement anomaly. In HFIR the gamma-ray spectrometer can be set up right next to the station where the grab sample is obtained, allowing measurement of the short-lived noble gases. This capability was demonstrated in the HRB-21 and NPR1/2 capsules where R/B data were reported for ^{85m}Kr, ^{87}Kr, ^{88}Kr, ^{89}Kr (3.15-min half-life), ^{133}Xe, ^{135}Xe, ^{135m}Xe, ^{137}Xe (3.82-m in half-life), and ^{138}Xe.

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Appendix C. DETAILED SAFETY TESTING AND PIE TASK LISTING

3.3 SAFETY TESTING AND PIE—GOALS, DESCRIPTION OF NEEDS, GAPS, AND R&D TASKS

The goal of the Safety Testing and PIE Data Needs task is to provide the postirradiation examination (PIE) techniques and safety testing capabilities that are suitable for the needs of the advanced gas reactor (AGR) program. In general, these capabilities will allow for the determination of the performance of the fuel kernel and coatings as a function of irradiation parameters and accident behavior under normal and chemical attack conditions. The actual programmatic needs for PIE measurements and safety tests will depend on the specific objectives of each irradiation test, the as-monitored online fission gas release-to-birth (R/B) measurements, and initial PIE observations. However, it is envisioned that several general tools will be routinely used that include dimensional measurements, gamma scanning, metallography, scanning electron microscope (SEM)/microprobe, chemical leaching, and heating to accident temperatures. The PIE techniques to be employed will generally be standard techniques with a proven track record; however, it is anticipated that old techniques may have to be improved and new ones developed if the program explores areas beyond those developed for past HTGR fuels work. Every effort will be made to produce data that are consistent with the historical database.

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 the data needs. Much of the needed facilities and apparatus are currently in place, although some modifications and upgrades will be necessary. In a few cases, the development of a new measurement technique may be required to satisfy a data need, leading to a research and development (R&D) task to develop or apply that new technique.

One caution is that the known PIE techniques vary in both the quantity and accuracy of the data that can be collected in a reasonable time at an affordable cost. Much PIE work is done remotely and can be expensive with limited accuracy. Thus, it may not be possible to satisfy areas of investigation that require a large number of highly accurate measurements in a reasonable time and at a reasonable cost. Data needs should be reviewed for their measurement needs and sensitivity to uncertainty before committing to a PIE task.

The major goals of the PIE and safety testing activity follow:

1. Collect relevant fuel PIE and accident testing data for the confirmation of normal performance, modeling needs, and behavior under accident conditions as a function of temperature, burnup, fast fluence, and chemical attack.
2. Cooperate with other DOE programs, and use international collaboration to the maximum extent possible for the resolution of key design and data needs and to minimize duplication of effort.

These are important assumptions of the activity:

1. AGRs will be designed such that the radionuclides are essentially retained in the core during normal operation and all design basis accidents.
2. TRISO fuel that meets fuel product specifications, in-core performance requirements, and accident behavior can be mass-produced at acceptable cost.
3. Radiologically significant reactivity transients are precluded by design. Thus, only a very limited amount of reactivity insertion accident (RIA) work needs to be done.
4. Water ingress accidents are moderate (~10,000 ppm H_2O) rather than core flooding.
5. Air and steam ingress accidents are to be considered.
6. The data can be collected within programmatic resources and within the schedule.

7. The data uncertainty will be sufficiently small to support the NRC-RES acceptance and eventual NRC-NRR licensing of the concept.
8. Small-scale laboratory testing is sufficient; full or scaled prototypes are not necessary, and the relevant tests can be conducted in a laboratory setting.
9. DOE will implement the requisite cooperative agreements to facilitate cooperation with other DOE programs as well as international cooperation.

These are necessary requirements for this activity:

1. Confirm fuel performance under normal and accident conditions to within the prescribed accuracy limits.
2. 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.
3. Rank (observed and measured) phenomena affecting the fuel performance/release.

The following subsections discuss activities required to prepare to conduct the PIE and safety testing activities, list the capsules with a brief summary of the PIE and safety testing objectives of each capsule, and then provide a detailed discussion of the PIE tasks with identification of the subset of tasks to be performed for each of the capsules, including cost and schedule estimates.

3.3.1 General PIE, Assessment, and Facility Preparation

The general PIE needs of the program involve capsule handling, capsule opening, fuel examination, fission product transport, fuel failure fraction determination, and accident testing. Much of this equipment is already in place, although some upgrading, improvements, and new capabilities may be necessary. Most of these tasks have been conducted in the past and present no unusual difficulties.

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 the capability of current facilities. Thus, an early task will be to determine the best way, within cost and schedule constraints, to conduct the PIE within the DOE complex.

The facilities and apparatus required to perform the PIE task must be made ready and in some cases upgraded to meet current performance expectations. While much of the HTGR fuels examination apparatus has remained operable over the years, minimal maintenance has been performed on this equipment, and little upgrading has been done. Specific tasks include the fabrication of jigs and fixtures, testing and replacement of machining tools, testing and inspection of HTGR-specific apparatus, upgrading software for the Core Conduction Cooldown Test Facility (CCCTF), checking CCCTF wiring and flow paths, preparation of environmental, safety, and health (ES&H) documentation, and review of operating procedures.

Generally, the nominal PIE task time is approximately 1 year, assuming that facilities and personnel are available. However, the AGR capsule irradiation schedule and potential PIE needs from other HTGR programs will result in multiple capsules undergoing PIE at the same time. Thus, expansion of the PIE capabilities at one site or the sharing of PIE work at two sites will be necessary to handle the workload. This need is most pressing for complex, time-consuming tasks such as high-temperature annealing. Thus, construction of an additional CCCTF and possibly an irradiated microsphere gamma analysis (IMGA) will be necessary to handle the workload within the schedule. The costs for duplication of an additional, special, one-of-a-kind apparatus have not been included in these cost estimates, and the program should reexamine the need for additional PIE infrastructure costs because multiple capsules require PIE services in the same time frame. This need will be addressed in Task 3.3.1.1, and the outcome of this task will influence the out-year PIE infrastructure costs.

3.3.1.1 PIE site task

Review the capabilities of candidate facilities, existing and new, for performing the separate PIE tasks, developing new equipment to perform new tasks not possible at present, and how these facilities might be integrated. Consider the implications of transport and time delays that might impact analysis, cost, and schedule. Also, consider PIE throughput and load-leveling issues (facility and staff). Provide necessary supporting information and participate in the process of selecting facilities to be used to conduct the PIE tasks. The number and timing of capsules will greatly impact this planning.

3.3.1.2 PIE preparation task

Prepare the PIE facility for the AGR PIE task. Design and fabricate jigs and fixtures for the handling of capsules, fuel forms, unbonded particles, and capsule internal components. Check cutting and machining tools for proper operation; replace or modify as necessary. Check the metallograph and the SEM/microprobe for proper operation with HTGR fuel, perform preventative maintenance, and any necessary calibrations. Inspect the IMGA for proper operation, and repair/replace components as necessary. Prepare the necessary glassware and furnace equipment for fuel form deconsolidation and leach-burn-leach. Review the CCCTF configuration, replace or repair components as necessary, and simplify the historical configuration. Upgrade the obsolete computer operating system. Document the configurations of the HTGR equipment, and review or rewrite the operating procedures. Inventory the PIE capability, and note equipment that needs to be developed outside of this task. Finally, prepare the necessary ES&H documentation for this task.

Important equipment deficiencies exist in the DOE complex for performing the required AGR tasks. Some of this results from the scrapping or obsolescence of equipment developed in the past, and some results from equipment that needs to be developed specifically for this program.

3.3.1.3 PGA equipment development task

Develop a particle gas analyzer (PGA) to crush a particle at a specified temperature and analyze the released gases. Of interest are CO_2, CO, isotopes of krypton, and isotopes of xenon. The possibility of making absolute pressure measurements should be investigated. A throughput of at least several particles per day is required. A device of this nature once existed, and the original design should be used as a starting point for the equipment development and the parameters to be measured.

3.3.1.4 Helium/air/steam CCCTF task

Develop the capability to work with air and steam ingress conditions at the temperatures of programmatic interest. A new fuel heating facility will be developed to extend the chemical environment capabilities beyond that of the current helium atmosphere furnace and to handle the increased workload. This facility and equipment development will require the investigation of new materials, examination of material transport to determine the concentration of the reactants at the fuel surface, gas monitoring equipment, and a system for the introduction and handling of air and steam both into and out of the furnace. In addition, the problem of collection of released fission products must be reexamined because air and steam may change the collection behavior of the coldfinger approach.

3.3.1.5 Coating physical properties equipment development task

Develop tools to investigate irradiated coating physical properties, especially the structure and anisotropic nature of carbon. Some material properties of interest might be strength, density, microstructure, layer bonding, permeability, and elastic modulus. Coordination with other groups that

perform material characterizations and fuel modeling will be conducted to define the areas of interest and develop the necessary techniques. In addition, the practicality of collecting these data to the required accuracy must be addressed. Further interaction with other program groups will be necessary to define this task, identify the key apparatus to be developed, and review its cost-effectiveness.

3.3.1.6 Fuel element reactivation equipment development task

Develop a method to reactivate a fuel element at a specific temperature so that its R/B can be measured. In the past, a King furnace in a TRIGA reactor was used. This design should serve as a starting point for the task (Argonne National Laboratory-West has a TRIGA reactor). In addition, a method of reactivation of a fuel element prior to its introduction into the CCCTF is necessary so that short-lived isotopes may be studied. A method of rapid transport between the reactor and the CCCTF will be required as well. The program should also review methods for detecting short-lived isotopes such as ^{131}I and ^{129}Te to be sure they can be accurately measured in the other PIE tasks. Alternately, a method of activating stable isotopes of iodine and tellurium may be investigated.

Figure C.1 shows a tentative cost and schedule for these activities.

3.3.2 AGR-1: PIE Shakedown Early Fuel Capsule

The first capsule to undergo irradiation and PIE is AGR-1. The purpose of this capsule is to gain experience with multicell capsule design, fabrication, and operation and to reduce chances of capsule or cell failures in subsequent capsules. It also provides early data on irradiated fuel performance and supports development of a fundamental understanding of the relationship between fuel fabrication process and fuel product properties and irradiation performance. Table C.1 identifies the tasks to be performed on this capsule, and Fig. C.2 shows the tentative cost and schedule. Note that the PIE options shown in Fig. C.2 are discussed in detail in Sect. 3.3.10.

3.3.3 AGR-2: PIE Performance Test Capsule

The purpose of the AGR-2 PIE is to provide irradiated fuel performance data beyond the R/B measurements for key fuel product/process variants to broaden options and increase prospects for meeting fuel performance requirements and to support development of a fundamental understanding of the relationship between the fuel fabrication process, fuel product properties, and irradiation performance. Table C.1 identifies the tasks to be performed on this capsule, and Fig. C.3 shows the tentative cost and schedule.

3.3.4 AGR-3: PIE Fission Product Transport

The purpose of the AGR-3 PIE is to collect data on fission product metal diffusion in kernels and coatings for use in development of fission product transport models. This PIE will focus on specially designed transport specimens rather than the fuel. Table C.1 identifies the tasks to be performed on this capsule, and Fig. C.4 shows the tentative cost and schedule. This PIE makes heavy use of the CCCTF, and additional capability in this area may be necessary.

3.3.5 AGR-4: PIE Fission Product Transport

The purpose of this PIE is to collect data on fission product diffusivities and sorptivities in graphite materials for use in the development of fission product transport models and overall codes. This PIE will focus on specially designed transport specimens rather than the fuel. Table C.1 identifies the tasks to be performed on this capsule, and Fig. C.5 shows the tentative costs and schedules.

ID	ID	Task Name	Duration	Cost
1	**1**	**General PIE, Assessment, and Preparation**	**840 days**	**$8,440,600.00**
2	2	3.3.1.1 PIE Site Task	90 days	$127,400.00
3	3	3.3.1.2 PIE Preparation Task	150 days	$636,700.00
4	4	3.3.1.3 PGA Equipment Development Task	250 days	$709,000.00
5	5	3.3.1.4 Helium/Air/Steam CCCTF Task	750 days	$5,225,000.00
6	6	3.3.1.5 Coating Physical Properties Equipment Development	250 days	$442,500.00
7	7	3.3.1.6 Fuel Element Re-Activatiion Equipment Developmnet Task	250 days	$1,300,000.00

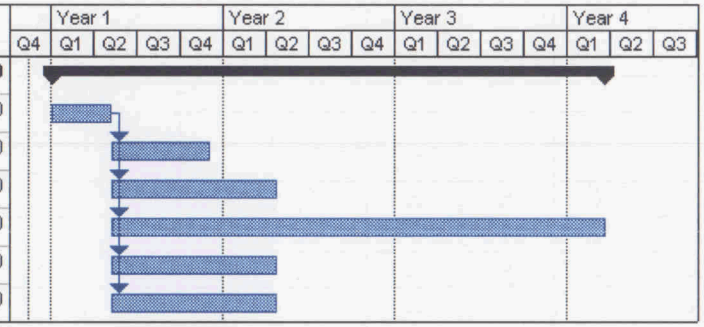

Fig. C.1. Tentative cost and schedule for the outlined PIE preparation task.

Table C.1. Capsule PIE tasks

Task No.	Task name	AGR-1	AGR-2	AGR-3	AGR-4	AGR-5	AGR-6	AGR-7	AGR-8
PIE TASK-1	Load Irradiation Capsule	X	X	X	X	X	X	X	X
PIE TASK-2	Capsule Gamma Scanning	X	X	X	X	X	X	X	X
PIE TASK-3	Capsule Opening	X	X	X	X	X	X	X	X
PIE TASK-4	Component Metrology	X	X	X	X	X	X	X	X
PIE TASK-5	Fuel Element Cross Section	X	X	X		X	X	X	X
PIE TASK-6	Fuel Element R/B Reactivation:		X	X				X	X
PIE TASK-7	Component Activity	X	X	X	X	X	X	X	X
PIE TASK-8	Leach-Burn-Leach	X	X			X	X	X	
PIE TASK-9	Fuel Element Deconsolidation	X	X					X	
PIE TASK-10	Irradiated Microsphere Gamma Analysis	X	X					X	
PIE TASK-11	Fuel Metallography	X	X			X	X	X	X
PIE TASK-12	Fuel Particle SEM Failure Mechanism	X	X					X	
PIE TASK-13	SEM Examination of Fission Products in Kernels and Coatings	X	X					X	
PIE TASK-14	Fission Gas and CO/CO_2 Content of Particle		X						
PIE TASK-15	Properties of Irradiated Materials Specimens	X	X						
PIE TASK-16	Radionuclide Transport in Irradiated Specimens		X		X				X
PIE TASK-17	Fission Product Release During Postirradiation Annealing	X		X	X	X	X	X	X
PIE TASK-18	Postannealing Metallography	X	X			X	X	X	
PIE TASK-19	Postannealing SEM	X	X			X	X	X	
PIE TASK-20	Waste Handling	X	X	X	X	X	X	X	X
PIE TASK-21	Reporting	X	X	X	X	X	X	X	X

ID	ID	Task Name	Duration	Cost
1	**1**	**AGR-1: PIE Shakedown Early Fuel Capsule**	**260 days**	**$1,774,090.00**
2	2	PIE Task-1 Load Irradiation Capsule Task	2 days	$18,840.00
3	3	PIE Task-2 Capsule Gamma Scanning	5 days	$23,500.00
4	4	PIE Task-3 Capsule Opening Task	25 days	$134,500.00
5	5	PIE Task-4 Component Metrology Task	15 days	$76,100.00
6	6	PIE Task-5 Fuel Element Cross-Section Task	15 days	$55,820.00
7	7	PIE Task-7 Component Activity Task	40 days	$58,000.00
8	8	PIE Task-8 Leach-Burn-Leach Task	5 days	$32,300.00
9	9	PIE Task-9 Fuel Element Deconsolidation Task	5 days	$26,300.00
10	10	PIE Task-10 IMGA	50 days	$135,900.00
11	11	PIE Task-11 Fuel Metallography Task	40 days	$131,560.00
12	12	PIE Task-12 Fuel Particle SEM Failure Mechanism Task	20 days	$95,320.00
13	13	PIE Task-13 SEM Examination of Fission Products in Kernel and Coatings Task	20 days	$47,660.00
14	14	PIE Task-15 Properties of Irradiated Materials Specimens Task	40 days	$63,500.00
15	15	PIE Task-17 Fission Release During Postirradiation Annealing Task	80 days	$464,400.00
16	16	PIE Task-18 Postannealing Metallography Task	15 days	$55,820.00
17	17	PIE Task-19 Postannealing SEM Task	20 days	$95,320.00
18	18	PIE Task-20 Waste Handling Task	260 days	$84,250.00
19	19	PIE Task-21 Reporting Task	260 days	$175,000.00

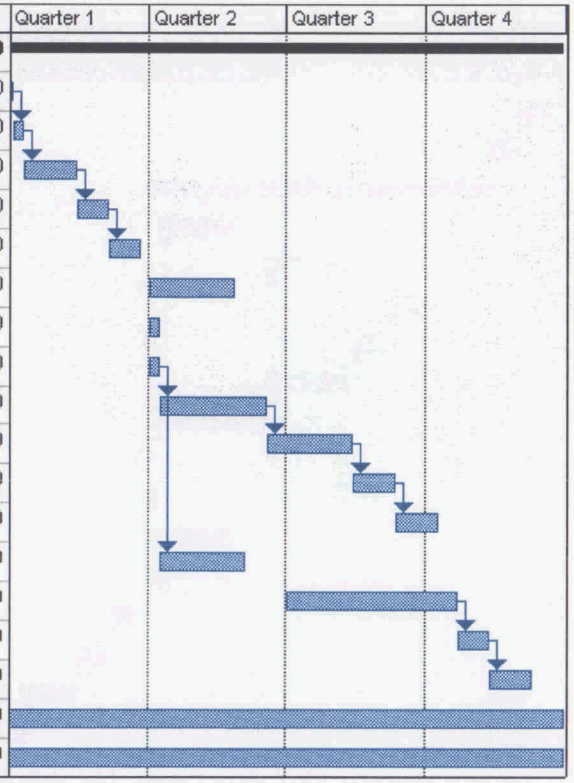

Fig. C.2. Capsule AGR-1 tentative cost and schedule.

ID	ID	Task Name	Duration	Cost
1	**1**	**AGR-2: PIE Performance Test Capsule**	**261 days**	**$2,623,350.00**
2	2	PIE Task-1 Load Irradiation Capsule Task	2 days	$18,840.00
3	3	PIE Task-2 Capsule Gamma Scanning	5 days	$23,500.00
4	4	PIE Task-3 Capsule Opening Task	25 days	$134,500.00
5	5	PIE Task-4 Component Metrology Task	15 days	$76,100.00
6	6	PIE Task-5 Fuel Element Cross-Section Task	15 days	$55,820.00
7	7	PIE Task-6 Fuel Element R/B Re-Activation Task	10 days	$105,400.00
8	8	PIE Task-7 Component Activity Task	40 days	$116,000.00
9	9	PIE Task-8 Leach-Burn-Leach Task	5 days	$32,300.00
10	10	PIE Task-9 Fuel Element Deconsolidation Task	5 days	$26,300.00
11	11	PIE Task-10 IMGA	50 days	$135,900.00
12	12	PIE Task-11 Fuel Metallography Task	40 days	$131,560.00
13	13	PIE Task-12 Fuel Particle SEM Failure Mechanism Task	20 days	$95,320.00
14	14	PIE Task-13 SEM Examination of Fission Products in Kernel and Coatings Task	20 days	$95,320.00
15	15	PIE Task-14 Fission Gas and CO/CO2 Content of Particles Task	30 days	$110,300.00
16	16	PIE Task-15 Properties of Irradiated Materials Specimens Task	40 days	$127,000.00
17	17	PIE Task-17 Fission Release During Postirradiation Annealing Task	160 days	$928,800.00
18	18	PIE Task-18 Postannealing Metallography Task	15 days	$55,820.00
19	19	PIE Task-19 Postannealing SEM Task	20 days	$95,320.00
20	20	PIE Task-20 Waste Handling Task	260 days	$84,250.00
21	21	PIE Task-21 Reporting Task	260 days	$175,000.00

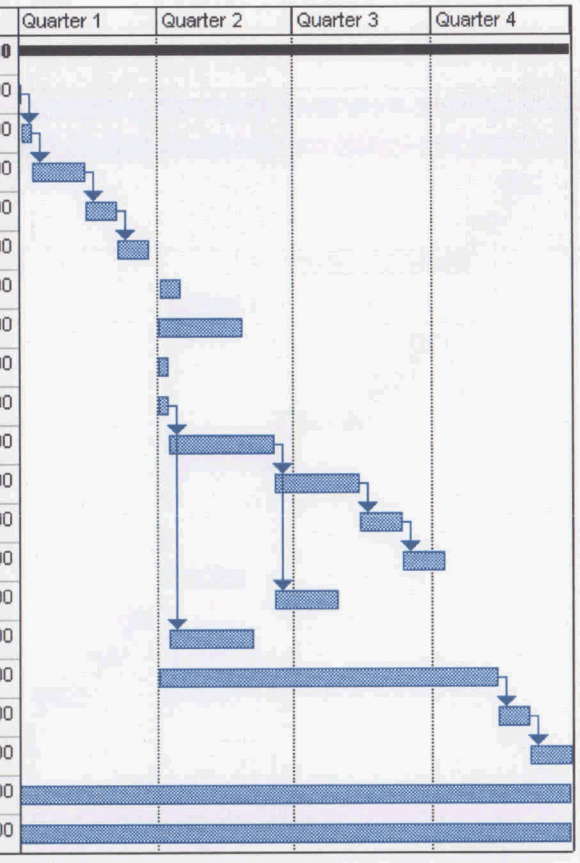

Fig. C.3. Capsule AGR-2 tentative cost and schedule.

ID	ID	Task Name	Duration	Cost
1	**1**	**AGR-3: PIE Fission Product Transport**	**261 days**	**$2,720,745.00**
2	2	PIE Task-1 Load Irradiation Capsule Task	2 days	$18,840.00
3	3	PIE Task-2 Capsule Gamma Scanning	5 days	$23,500.00
4	4	PIE Task-3 Capsule Opening Task	25 days	$134,500.00
5	5	PIE Task-4 Component Metrology Task	15 days	$76,100.00
6	6	PIE Task-5 Fuel Element Cross-Section Task	15 days	$13,955.00
7	7	PIE Task-6 Fuel Element R/B Re-Activation Task	10 days	$105,400.00
8	8	PIE Task-7 Component Activity Task	214 days	$696,000.00
9	9	PIE Task-17 Fission Release During Postirradiation Annealing Task	214 days	$1,393,200.00
10	10	PIE Task-20 Waste Handling Task	260 days	$84,250.00
11	11	PIE Task-21 Reporting Task	260 days	$175,000.00

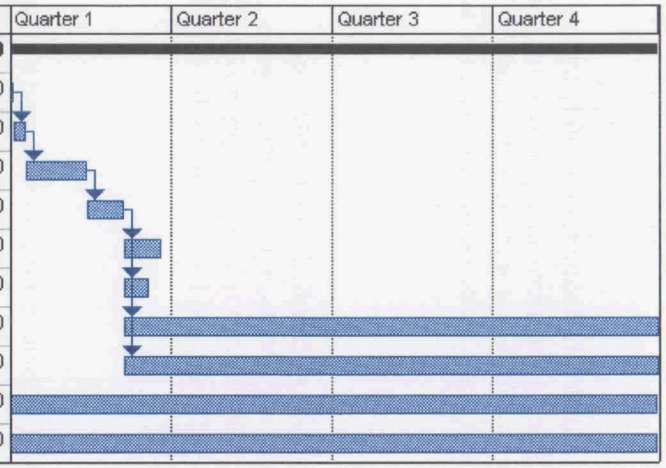

Fig. C.4. Capsule AGR-3 tentative cost and schedule.

ID	ID	Task Name	Duration	Cost
1	**1**	**AGR-4: PIE Fission Product Transport**	**261 days**	**$2,030,090.00**
2	2	PIE Task-1 Load Irradiation Capsule Task	2 days	$18,840.00
3	3	PIE Task-2 Capsule Gamma Scanning	5 days	$23,500.00
4	4	PIE Task-3 Capsule Opening Task	25 days	$134,500.00
5	5	PIE Task-4 Component Metrology Task	15 days	$76,100.00
6	6	PIE Task-7 Component Activity Task	40 days	$116,000.00
7	7	PIE Task-16 Radionuclide Transport in Irradiated Specimens Task	174 days	$937,500.00
8	8	PIE Task-17 Fission Release During Postirradiation Annealing Task	80 days	$464,400.00
9	9	PIE Task-20 Waste Handling Task	260 days	$84,250.00
10	10	PIE Task-21 Reporting Task	260 days	$175,000.00

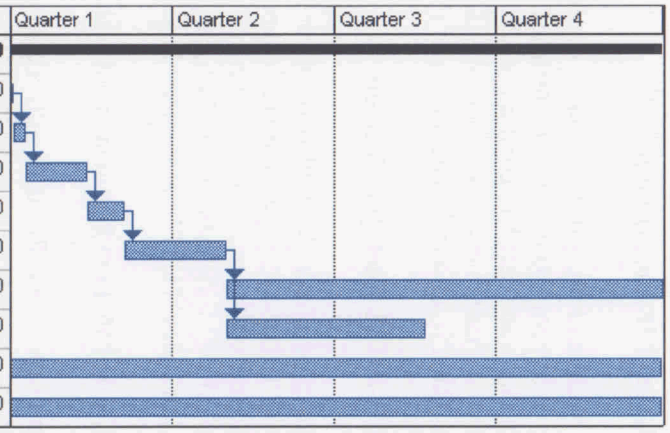

Fig. C.5. Capsule AGR-4 tentative cost and schedule.

3.3.6 AGR-5: PIE Fuel Qualification

The purpose of this PIE is to collect irradiation 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. The primary interest is verifying proper fuel performance. Table C.1 identifies the tasks to be performed on this capsule, and Fig. C.6 shows the tentative costs and schedules. This PIE also makes heavy use of the CCCTF.

3.3.7 AGR-6: PIE Fuel Qualification

The purpose of this PIE is also to collect irradiation 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. The primary interest is verifying proper fuel performance. Table C.1 identifies the tasks to be performed on this capsule, and Fig. C.7 shows the tentative costs and schedules. This PIE also makes heavy use of the CCCTF.

3.3.8 AGR-7: PIE Fuel Performance Limits Capsule

The purpose of this PIE is to collect data on the capability of the selected fuel to withstand irradiation and accidents conditions beyond the conditions in capsules AGR-5 and -6 in support of plant design and licensing. Table C.1 identifies the tasks to be performed on this capsule, and Fig. C.8 shows the tentative costs and schedules.

3.3.9 AGR-8: PIE Fission Product Transport

The purpose of this PIE is to collect irradiated fuel performance data on fission product metal releases from compacts with known quantities of failed particles for use in validation of fission product transport codes. Table C.1 identifies the tasks to be performed on this capsule, and Fig. C.9 shows the tentative costs and schedules. This PIE also makes heavy use of the CCCTF.

3.3.10 Detailed PIE Scope of Activities

A capsule PIE 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 serially. For example, a capsule must be opened before any work can be done with the fuel, so it is a serial task. Fuel element 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, unrelated tasks. 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 a PIE will take approximately 1 year to complete with no restrictions on resources. The following tasks outline the options that are likely to be available for a particular PIE. The actual tasks that will be performed for a particular capsule are shown in Table C.1.

The PIE task will have to be integrated with other AGR groups so that the tasks can be conducted in an efficient and time-effective manner. Table C.2 shows the likely interfaces between groups. The primary goal is to ensure that the needed 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.

Table C.3 identifies the deliverable to be produced in the course of conducting the PIE activities, while Table C.4 summarizes the capsule PIE objectives and estimated cost. These capsules are in an early stage of definition, but their requirements show the variation in the costs as the scope changes. For

ID	ID	Task Name	Duration	Cost
1	**1**	**AGR-5: PIE Fuel Qualification**	**261 days**	**$2,962,240.00**
2	2	PIE Task-1 Load Irradiation Capsule Task	2 days	$18,840.00
3	3	PIE Task-2 Capsule Gamma Scanning	5 days	$23,500.00
4	4	PIE Task-3 Capsule Opening Task	25 days	$134,500.00
5	5	PIE Task-4 Component Metrology Task	15 days	$76,100.00
6	6	PIE Task-5 Fuel Element Cross-Section Task	15 days	$55,820.00
7	7	PIE Task-7 Component Activity Task	40 days	$29,000.00
8	8	PIE Task-8 Leach-Burn-Leach Task	5 days	$32,300.00
9	9	PIE Task-11 Fuel Metallography Task	40 days	$32,890.00
10	10	PIE Task-17 Fission Release During Postirradiation Annealing	163 days	$1,393,200.00
11	11	PIE Task-18 Postannealing Metallography Task	173 days	$334,920.00
12	12	PIE Task-19 Postannealing SEM Task	131 days	$571,920.00
13	13	PIE Task-20 Waste Handling Task	260 days	$84,250.00
14	14	PIE Task-21 Reporting Task	260 days	$175,000.00

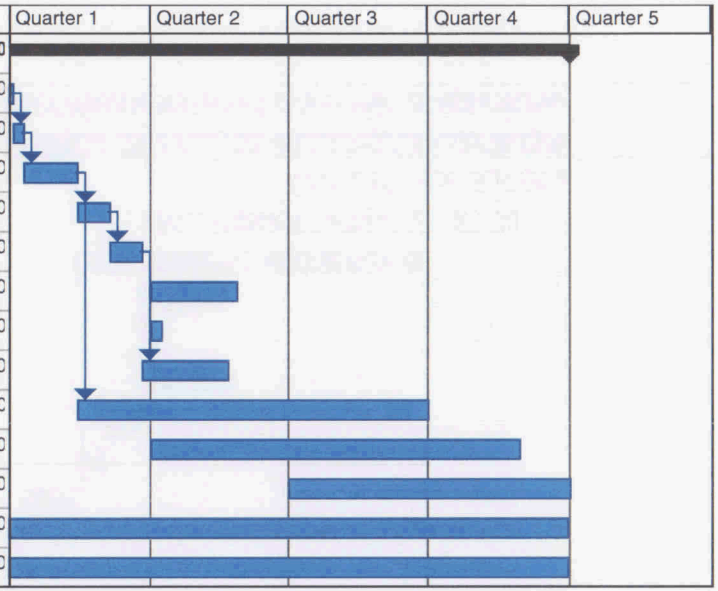

Fig. C.6. Capsule AGR-5 tentative cost and schedule.

ID	ID	Task Name	Duration	Cost
1	**1**	**AGR-6: PIE Fuel Qualification**	**261 days**	**$2,962,240.00**
2	2	PIE Task-1 Load Irradiation Capsule Task	2 days	$18,840.00
3	3	PIE Task-2 Capsule Gamma Scanning	5 days	$23,500.00
4	4	PIE Task-3 Capsule Opening Task	25 days	$134,500.00
5	5	PIE Task-4 Component Metrology Task	15 days	$76,100.00
6	6	PIE Task-5 Fuel Element Cross-Section Task	15 days	$55,820.00
7	7	PIE Task-7 Component Activity Task	40 days	$29,000.00
8	8	PIE Task-8 Leach-Burn-Leach Task	5 days	$32,300.00
9	9	PIE Task-11 Fuel Metallography Task	40 days	$32,890.00
10	10	PIE Task-17 Fission Release During Postirradiation Annealing	163 days	$1,393,200.00
11	11	PIE Task-18 Postannealing Metallography Task	173 days	$334,920.00
12	12	PIE Task-19 Postannealing SEM Task	131 days	$571,920.00
13	13	PIE Task-20 Waste Handling Task	260 days	$84,250.00
14	14	PIE Task-21 Reporting Task	260 days	$175,000.00

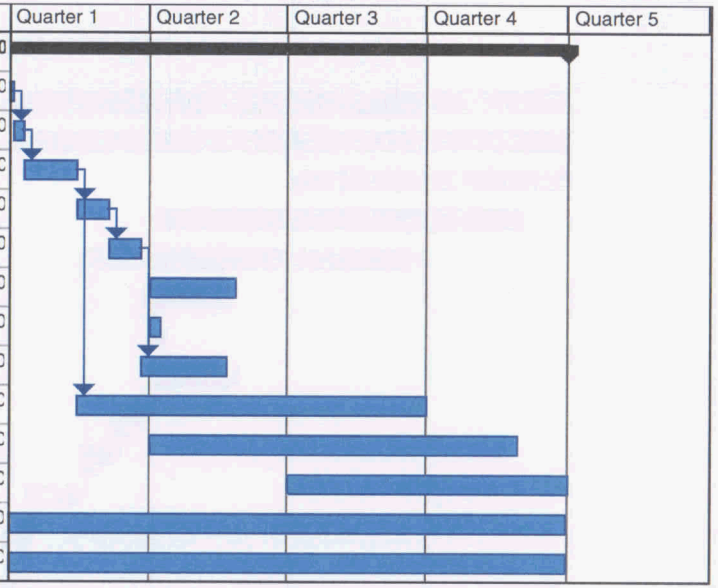

Fig. C.7. Capsule AGR-6 tentative cost and schedule.

ID	ID	Task Name	Duration	Cost
1	**1**	**AGR-7: PIE Fuel Performance Limits Capsule**	**261 days**	**$3,519,150.00**
2	2	PIE Task-1 Load Irradiation Capsule Task	2 days	$18,840.00
3	3	PIE Task-2 Capsule Gamma Scanning	5 days	$23,500.00
4	4	PIE Task-3 Capsule Opening Task	25 days	$134,500.00
5	5	PIE Task-4 Component Metrology Task	15 days	$76,100.00
6	6	PIE Task-5 Fuel Element Cross-Section Task	15 days	$55,820.00
7	7	PIE Task-6 Fuel Element R/B Re-Activation Task	10 days	$105,400.00
8	8	PIE Task-7 Component Activity Task	40 days	$29,000.00
9	9	PIE Task-8 Leach-Burn-Leach Task	5 days	$32,300.00
10	10	PIE Task-9 Fuel Element Deconsolidation Task	5 days	$26,300.00
11	11	PIE Task-10 IMGA	50 days	$135,900.00
12	12	PIE Task-11 Fuel Metallography Task	40 days	$131,560.00
13	13	PIE Task-12 Fuel Particle SEM Failure Mechanism Task	20 days	$95,320.00
14	14	PIE Task-13 SEM Examination of Fission Products in Kernel and Coatings Task	20 days	$95,320.00
15	15	PIE Task-17 Fission Release During Postirradiation Annealing Task	163 days	$1,393,200.00
16	16	PIE Task-18 Postannealing Metallography Task	173 days	$334,920.00
17	17	PIE Task-19 Postannealing SEM Task	131 days	$571,920.00
18	18	PIE Task-20 Waste Handling Task	260 days	$84,250.00
19	19	PIE Task-21 Reporting Task	260 days	$175,000.00

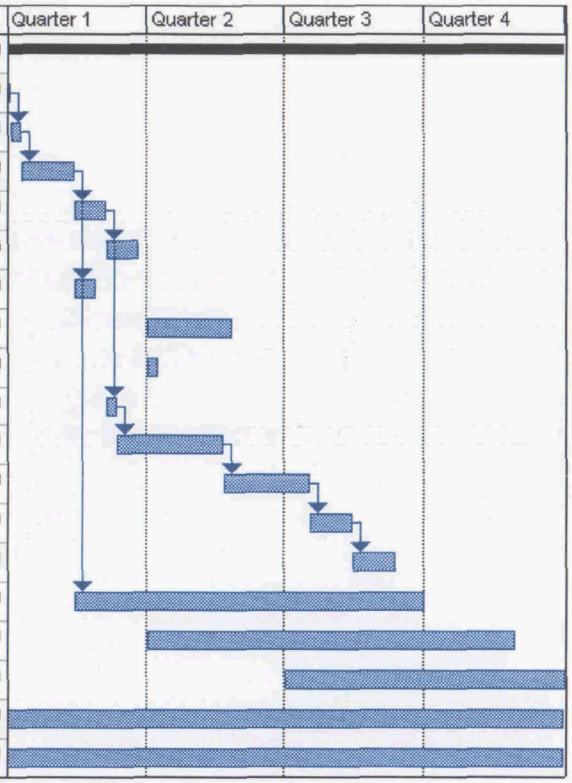

Fig. C.8. Capsule AGR-7 tentative cost and schedule.

ID	ID	Task Name	Duration	Cost
1	**1**	**AGR-8: PIE Fission Product Transport**	**261 days**	**$2,445,235.00**
2	2	PIE Task-1 Load Irradiation Capsule Task	2 days	$18,840.00
3	3	PIE Task-2 Capsule Gamma Scanning	5 days	$23,500.00
4	4	PIE Task-3 Capsule Opening Task	25 days	$134,500.00
5	5	PIE Task-4 Component Metrology Task	15 days	$76,100.00
6	6	PIE Task-5 Fuel Element Cross-Section Task	15 days	$13,955.00
7	7	PIE Task-6 Fuel Element R/B Re-Activation Task	10 days	$105,400.00
8	8	PIE Task-7 Component Activity Task	214 days	$696,000.00
9	9	PIE Task-11 Fuel Metallography Task	40 days	$32,890.00
10	10	PIE Task-16 Radionuclide Transport in Irradiated Specimens Task	214 days	$156,000.00
11	11	PIE Task-17 Fission Release During Postirradiation Annealing Task	214 days	$928,800.00
12	12	PIE Task-20 Waste Handling Task	260 days	$84,250.00
13	13	PIE Task-21 Reporting Task	260 days	$175,000.00

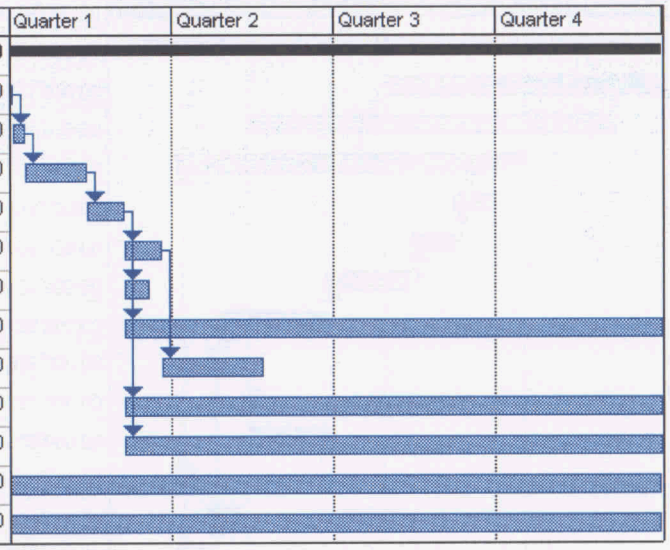

Fig. C.9. Capsule AGR-8 tentative cost and schedule.

Table C.2. Group interface matrix

Task No.	Task name	Fuel and materials irradiation	Fuel manufacture	Fission product and source term	Fuel performance and modeling	Comments
3.3.1.1	PIE Site	X		X	X	Interest in capabilities at each site and whether they will meet needs
3.3.1.2	PIE Preparation	X	X	X	X	Equipment to be used and its status
3.3.1.3	PGA Equipment Development	X			X	Kernel gas release, coating stresses, and oxygen potential
3.3.1.4	Air/Steam CCCTF Modification	X			X	Need to complete accident testing under chemical attack. Reactant concentrations need better definition.
3.3.1.5	Coating Physical Properties Equipment Development	X	X		X	Collect data for models; a wide range of properties to be measured. Several facilities may be involved.
3.3.1.6	Fuel Element Reactivation Equipment Development	X		X	X	Need for short-lived isotopes and individual R/B measurements. Also should investigate ^{131}I and ^{129}Te measurement techniques. A backup would be activating stable isotopes.
PIE TASK-1	Load Irradiation Capsule	X				Coordinate with irradiation group for shipping and paperwork
PIE TASK-2	Capsule Gamma Scanning	X				First check of irradiation integrity and fuel element status
PIE TASK-3	Capsule Opening	X		X		Recovery of fuel and capsule items; internal visual status
PIE TASK-4	Component Metrology	X	X			Indications of dimensional stability under irradiation

Table C.2. (continued)

Task No.	Task name	Fuel and materials irradiation	Fuel manufacture	Fission product and source term	Fuel performance and modeling	Comments
PIE TASK-5	Fuel Element Cross Section	X	X		X	First indication of fuel performance (after R/B). Gross indication of performance
PIE TASK-6	Fuel Element R/B Reactivation	X		X		Sort fuel elements by R/B, so good/bad compacts can be segregated
PIE TASK-7	Component Activity	X		X		Transport data? Will this be useful to the program?
PIE TASK-8	Leach-Burn-Leach	X	X	X	X	First measurement of fuel quality based on metals as well as gases
PIE TASK-9	Fuel Element Deconsolidation	X	X	X	X	Source of particles for individual examination
PIE TASK-10	Irradiated Microsphere Gamma Analysis	X		X	X	Find bad particles, determine absolute inventory, determine burnup, and estimate burnup gradients
PIE TASK-11	Fuel Metallography	X	X	X	X	Begin detailed examination of particles and coating behavior
PIE TASK-12	Fuel Particle SEM Failure Mechanism	X	X		X	Detailed check of coating behavior and fission product migration
PIE TASK-13	SEM Examination of Fission Products in Kernels and Coatings			X		Diffusion data and fission product migration
PIE TASK-14	Fission Gas and CO/CO_2 Content of Particle	X	X		X	Need for coating stress calculations, kernel release estimations, and oxygen potential

Table C.2. (continued)

Task No.	Task name	Fuel and materials irradiation	Fuel manufacture	Fission product and source term	Fuel performance and modeling	Comments
PIE TASK-15	Properties of Irradiated Materials Specimens	X	X		X	Need for coating stress calculations, material changes under irradiation, particle stability estimations under irradiation
PIE TASK-16	Radionuclide Transport in Irradiated Specimens			X		Transport data and diffusion coefficients
PIE TASK-17	Fission Product Release During Postirradiation Annealing	X	X	X	X	First indications of accident behavior, high-temperature diffusion coefficients, and thermal changes in materials
PIE TASK-18	Postannealing Metallography	X			X	Material behavior for accidents and thermal material changes
PIE TASK-19	Postannealing SEM	X		X	X	Material behavior for accidents, transport information, and fission product chemical attack
PIE TASK-20	Waste Handling	N/A				Routine activities
PIE TASK-21	Reporting					

Table C.3. Deliverables

Task No.	Task name	Equipment development plan	Experimental test plan	Task or equipment documentation report	Quick-look report	PIE final report
3.3.1.1	PIE Site			X	Items of immediate program importance will be selected and reported on in part	
3.3.1.2	PIE Preparation			X		
3.3.1.3	PGA Equipment Development	X		X		
3.3.1.4	Air/Steam CCCTF Modification	X		X		
3.3.1.5	Coating Physical Properties Equipment Development	X		X		
3.3.1.6	Fuel Element Re-Activation Equipment Development	X		X		
PIE TASK-1	Load Irradiation Capsule		PIE tasks will be selected and detailed in the test plan			PIE tasks will be detailed and results reported in the final report
PIE TASK-2	Capsule Gamma Scanning					
PIE TASK-3	Capsule Opening					
PIE TASK-4	Component Metrology					
PIE TASK-5	Fuel Element Cross Section					
PIE TASK-6	Fuel Element R/B Re-Activation:					
PIE TASK-7	Component Activity					
PIE TASK-8	Leach-Burn-Leach					
PIE TASK-9	Fuel Element Deconsolidation					
PIE TASK-10	Irradiated Microsphere Gamma Analysis					
PIE TASK-11	Fuel Metallography					
PIE TASK-12	Fuel Particle SEM Failure Mechanism					
PIE TASK-13	SEM Examination of Fission Products in Kernels and Coatings					
PIE TASK-14	Fission Gas and CO/CO_2 Content of Particle					
PIE TASK-15	Properties of Irradiated Materials Specimens					
PIE TASK-16	Radionuclide Transport in Irradiated Specimens					
PIE TASK-17	Fission Product Release During Postirradiation Annealing					
PIE TASK-18	Postannealing Metallography					
PIE TASK-19	Postannealing SEM					
PIE TASK-20	Waste Handling	N/A				

Table C.4. Estimated PIE costs for the proposed AGR series capsules

Capsule	Objective	Cost ($M)
AGR-1	Gain experience with multicell capsule design, fabrication, and operation and reduce chances of capsule or cell failures in subsequent capsules; early data on irradiated fuel performance support development of a fundamental understanding of the relationship between fuel fabrication process and fuel product properties and irradiation performance	1.8
AGR-2	Provide irradiated fuel performance data for key fuel product/process variants to broaden options and increase prospects for meeting fuel performance requirements and to support development of a fundamental understanding of the relationship between fuel fabrication process and fuel product properties and irradiation performance	2.6
AGR-3	Provide irradiated fuel performance data on fission product gas release from failed particles, fission product metal diffusion in kernels, and gas and metal diffusion in coatings for use in development of fission product transport models	2.7
AGR-4	Provide data on fission product diffusivities and sorptivities in graphite materials for use in development of fission product transport models	2.0
AGR-5	Provide irradiation 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	3.0
AGR-6	Provide irradiation 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	3.0
AGR-7	Provide data on capability of fuel to withstand conditions beyond AGR-5 and -6 in support of plant design and licensing	3.5
AGR-8	Provide irradiated fuel performance data on fission product gas and metal releases from compacts with known quantities of failed particles for use in validation of fission product transport codes	2.4

Note: Each PIE is assumed to require approximately 1 year. Individual tasks will be run in parallel to the extent possible to compress the schedule. If a serial task effort is required because of a specific program developmental need, the schedule would stretch out by approximately 6 to 12 months.

example, AGR-1 is a shakedown capsule that may provide early fuel performance information, while AGR-7 is a fuel qualification capsule that explores the fuel limits. The aggressive PIE schedule (two or more extensive PIEs per year) might exceed the current HTGR PIE facility handling limits and require infrastructure expansion.

The first step in the PIE process is to transfer the capsule from the irradiation facility to the PIE facility. The capsules are usually shipped by truck in a shielded cask. The process is considered to be routine, but preparations can be considerable and must be well planned.

PIE TASK-1: Load Irradiation Capsule: Complete the transfer and nuclear accountability documentation, and prepare the hot cell for the delivery of the cask. Preparation and shipping of the cask from the reactor to the hot cell is the responsibility of the irradiation group. Load the capsule into the hot cell, prepare the cask for return shipment, and ship the cask back. This is an established task.

Prior to opening the capsule, it will be gamma-scanned to check its internal integrity and, possibly, to determine if any fuel elements have broken or if significant amounts of fission products have been released and migrated within the capsule.

PIE TASK-2: Capsule Gamma Scanning: Prepare the capsule for gamma scanning, and gamma-scan the capsule. Produce a false color image of the capsule and any regions that appear abnormal. Note the integrity of the fuel elements. Rescan selected regions of the capsule for specific fission products if abnormal regions appear. This is an established task.

Using remote machine tools, the capsule will be opened, and the fuel and internal components of interest removed. They will be placed in labeled containers. If of interest, the capsule and other internal components will be segmented for later gamma-counting and leaching.

PIE TASK-3: Capsule Opening: Using in-cell machine tools and jigs, open the irradiation capsule and remove the fuel elements and internal components of experimental value. Place each item in a labeled container, and log its location. The irradiation capsule itself may be cut up into pieces if abnormal conditions are noted. Clean up the metal chips and any other debris for later waste disposal. This is an established task, but the complexity of the task depends on the specific capsule design and the required accuracy of the cutting.

The fuel and capsule components will be dimensionally and visually inspected for irradiation-induced changes. These data will be compared with preirradiation values and appearances.

PIE TASK-4: Component Metrology: Visually and dimensionally inspect the fuel elements and capsule internal components. The reference method uses mechanical tools such as dial indicators and V blocks, but other methods may be employed as time permits and if cost-effective. Compare appearances before and after irradiation; dimensionally inspect to program specified tolerances and compare with pre-irradiation values. If transfer of contamination is an issue (pickup from the cell or tools), this task can be done after other more sensitive tasks. In particular, the nonfuel items may be handled within their own task. This is an established task, but it may require specialized jigs and fixtures or even the development of special tools if a large number of measurements is necessary.

PIE TASK-5: Fuel Element Cross-Section: Examine cross sections of a fuel element by optical metallography to document conditions within the fuel element, including fuel particles and matrix. The examination will visually document conditions within fuel particles such as kernel migration, kernel porosity, buffer integrity, and the integrity of the TRISO layers (IPyC, OpyC, and SiC). If fuel failure fractions on the order of 1% are encountered, it may not be necessary to perform the leach-burn-leach technique to measure the failure fraction. This is an established technique.

PIE TASK-6: Fuel Element R/B Reactivation: Place fuel elements, one at a time, in a TRIGA or TRIGA-like reactor with an internal temperature-controlled furnace. Reirradiate the elements, and measure their R/B as a function of temperature. This task will allow the individual measurement of fuel element R/B (rather the capsule total R/B) and the identification of fuel elements with damaged fuel particles. The isolated fuel elements can then be segregated for further investigation.

A related task is to gamma-count capsule components (nontransport study items) or leach and gamma-count capsule components to determine the identity, migration, and sources of fission products. This task may or may not be necessary depending on the goal of a particular irradiation capsule.

PIE TASK-7: Component Activity: Individually gamma-count capsule components to determine the isotopes and amount of fission products present. The isotopes ^{110m}Ag and ^{137}Cs are typically of particular interest; but the experimental plan will detail the desired isotopes. If the activity of a base component is high (e.g., Inconel), the component may have to be leached to remove the fission products for an accurate counting. This task will concentrate on specific nonfuel items for general analysis. Graphite components may be routinely counted because they are good collectors of fission products and are near the fuel. Capsule parts may be leached if irradiation problems arise. Items designed specifically for fission product transport studies will be handled within their own task. This is an established technique.

For irradiations of fuel elements (compacts or pebbles), there will be a need to measure the fuel particle failure fraction independent of the online R/B measurements, because of the uncertainty in the

R/B for a single particle failure and the inability to measure metallic releases. The most useful technique to measure the fuel particle failure fraction is leach-burn-leach when the online R/B measurements suggest a failure fraction well under 1%. For failure fractions in the neighborhood of 1% or higher, metallography may aid in the determination of the failure fraction. For intermediate failure fractions of 10^{-4} to 10^{-2}, deconsolidation followed by IMGA may be useful.

PIE TASK-8: Leach-Burn-Leach: Measure fuel particle failure fraction in irradiated fuel elements using the leach-burn-leach technique. Consult with the program on leach and burn times and temperatures to avoid failures caused by too aggressive a test. Document the actual process, so long-term consistency can be assured. Exposed kernels and failed SiC bounded by intact OPyC are measured by this technique. The technique may not be reliable for failed SiC and intact IPyC (may not leach out uranium, but should see mobile fissions products like cesium) because of inadequate O_2 transport if the flaw is very minor. This is an established technique, but care must be taken to establish the test parameters.

To document the failure mechanism of fuel particles in fuel elements with a failure fraction less than about 1%, it will be necessary to isolate a number of failed fuel particles for examination. This will be done by fuel element deconsolidation to obtain individual particles, which are then put through the IMGA diagnostic system to identify and collect failed particles as well as determining their inventory of gamma-emitting fission products. Failed fuel particles will be first visually examined (nondestructive) and then examined by metallography (destructive) to observe failures in the TRISO layers. Some of these particles will be further examined by SEM and microprobe to look more closely at the nature of the layer failures using wavelength dispersive X-ray spectroscopy (WDX) to determine the influence of fission product elements or chemical contamination on the failure mechanism.

PIE TASK-9: Fuel Element Deconsolidation: Deconsolidate fuel elements by an electrochemical technique to obtain individual fuel particles; sieve particles to remove debris and wash and dry. Note condition of fuel. Analyze solution for uranium and fission products. Document the parameters used for long-term consistency. This is an established technique.

PIE TASK-10: Irradiated Microsphere Gamma Analysis: Gamma-scan a statistically significant number of particles to determine their fission product inventory, and identify and collect failed fuel particles. Compare mobile and nonmobile fission products such as ^{144}Ce and ^{137}Cs. Visually examine failed and suspect particles; photograph any unusual observations. This is an established technique; however, it is quite time-consuming, and operational problems can arise if the deconsolidation leaves much debris on the particles. This task may not be practical if the failure rate is very low ($<10^{-4}$).

PIE TASK-11: Fuel Metallography: Examine both good and failed fuel particles to document failures in TRISO coatings using optical metallography. Individually mount failed particles, and grind and polish them using traditional methods. To save time, good particles may be mounted as a batch and ground and polished as a group. This is an established technique; however, care must be taken because the working medium (usually water) may dissolve some fission products and the polishing may smear others over the face of the mount, complicating the interpretation of later SEM/microprobe investigations. Also, the probability of finding a particular coating flaw may be low if the flaw is very local.

PIE TASK-12: Fuel Particle SEM Failure Mechanism: Examine failed fuel particles by SEM/microprobe using WDX to elucidate the failure mechanism, and map the isotopes of interest. Isotopes of interest include, U, Pu, O, C, and the fission products. The relative presence of these elements in suspicious regions of the failed particle is of interest. This is an established technique; but the region examined by the SEM/microprobe is small, so the probability of finding a small local flaw may be low.

The distribution of fission products in kernels and within coatings is needed for validation of models of the diffusivity of fission products within the components of the TRISO fuel particle (i.e., kernel, buffer, IPyC, SiC, and OPyC) and the solubility within the PyC and SiC layers.

PIE TASK-13: Examination of Fission Products in Kernels and Coatings: Examine by SEM/microprobe (using WDX) the components of intact TRISO fuel particles to measure fission product contents (mapping) and concentration gradients within the kernel, buffer, IPyC, SiC, and OPyC. Fission products such as Ag, Cs, Sr, Te, and I are of particular interest. This is an established technique;

however, the accuracy of this task must be reviewed to be consistent with the required data. It may be difficult to get the required spatial resolution. The high noise level of radioactive specimens limits the detection limit of the SEM/microprobe. It may not be possible to find isotopes at small concentrations (less than a few percent). Also, the mount preparation process may smear fission products on the surface of the mount (grinding and polishing) or remove them by dissolving (water is often used in mount preparation). Gamma-counting pieces of coatings removed from a particle may be an alternative to this task.

The fission gas and CO contents of irradiated fuel kernels are needed to validate fuel performance models that calculate these quantities to predict gas loading in the TRISO fuel particles, to estimate kernel oxygen potential, and to estimate kernel gas release.

PIE TASK-14: Fission Gas and CO/CO_2 Content of Particle: Measure fission gas, CO_2, and CO contents of intact irradiated particles by mechanically breaking particles and collecting and analyzing the gases released. A gas chromograph or mass spectrometer will be used for this analysis. This once was an established technique, known as the PGA, but it is no longer in existence. New equipment will have to be designed, fabricated, and tested (Task 3.3.1.3 discussed previously).

Specific nonfuel items may be placed within an irradiation capsule for material property studies and fission product transport studies. The relationship between these items and the fuel elements is unknown at this time and may or may not involve special interactions.

Samples of materials, such as pyrocarbons, SiC, and graphite will be irradiated for the purpose of producing samples from which irradiation effects on material properties can be measured. These data are needed for development and validation of models for fuel performance.

PIE TASK-15: Properties of Irradiated Materials Specimens: Measure properties (e.g., thermal, physical, mechanical) as requested on samples of irradiated materials such as PyC, SiC, graphite, and metals. Some of this task may be done in other facilities (rather than the PIE facility) because the specimens may have sufficiently low levels of radionuclide content. Advantage will be taken of existing equipment in a variety of facilities and techniques developed elsewhere. Some needed measurement techniques may not be well established and may require development. This task will require further definition and development as the program proceeds.

Samples of materials, such as pyrocarbons, SiC, and graphite will be exposed to fission products from failed fuel particles during irradiation for the purpose of producing samples from which fission product solubility and diffusivity can be measured after irradiation. These data are needed for development and validation of models of fission product transport.

PIE TASK-16: Radionuclide Transport in Irradiated Specimens: Measure radionuclide content and gradients in irradiated specimens by appropriate established techniques, such as beta and gamma spectrometry and neutron activation. Special handling and segmenting of the specimens may be necessary to avoid contamination and perform the measurement. This task will require more definition and development as the program proceeds. Spatial resolution may be difficult to obtain.

An important goal of this program is to determine the performance of the fuel under high-temperature accident conditions. In particular, three environments are of interest: helium, and gas mixtures representative of air and steam ingress events. The fuel will be exposed to these environments for up to 500 h. The exact composition of these environments are not known at present, but it is assumed that the test will be run at atmospheric pressure.

The maximum temperature predicted for a core conduction cooldown accident in design basis analyses for AGRs is typically ~1600°C for about 25–50 h with a long (hundreds of hours) cooldown following. Traditionally, postirradiation isothermal annealing at temperatures of 1600°C, 1700°C, and 1800°C have been performed for several hundred hours with continuous collection of released fission products. Isothermal tests are generally considered to be conservative relative to heatup transient simulation tests, which follow more closely the time-temperature profiles calculated to occur in a core conduction cooldown transient, because more time is spent at the highest temperatures. However, this conclusion was challenged by the Nuclear Regulatory Commission (NRC) in the Exelon PBMR preapplication interactions based on the NRC's interpretation of German data. Thermal gradients are not

expected to be significant in either isothermal or transient simulation tests. Isothermal tests are easier to analyze than transient tests because time at temperature is the only variable during the test. Isothermal tests, transient simulation tests, or both will be conducted based on the needs of plant design and licensing as determined prior to the test. The data needed from safety testing are fission product release, TRISO coating layer integrity, and fission product distribution within fuel particles (corrosion likelihood) and fuel elements.

PIE TASK-17: Fission Product Release During Postirradiation Annealing: Conduct postirradiation isothermal annealing tests that measure fission product release as a function of time at temperatures in the range of 1400°C –1800°C, as requested by the AGR program. These safety tests can be performed on fuel elements or unbonded fuel particles. Three atmospheres are to be considered as possible testing options: helium, and gas mixtures representative of air and steam ingress events. Tests performed in helium will be conducted at approximately atmospheric pressure with high-purity helium. A special coldfinger airlock arrangement within the annealing furnace will allow the collection of metallic fission products at discrete intervals. A continuous gas monitoring system will collect and analyze the release of ^{85}Kr. This is an established technique. Testing in air-gas mixtures representative of air and steam ingress events is not currently available, and special equipment would have to be developed for this program if these tests are needed. The collection of metallic fission products may not be as straightforward as in the helium atmosphere. If individual particles are heated, the IMGA system may be used to measure the before and after heating inventories, which allows precise determination of releases and identification of the releasing particles.

PIE TASK-18: Postannealing Metallography: Characterize TRISO-coating layer integrity by visual metallography, looking for evidence of SiC layer thinning and decomposition, chemical attack of SiC, and mechanical condition and microstructures of the SiC and PyC layers. This is an established technique.

PIE TASK-19: Postannealing SEM: Measure (map) fission product distribution (especially Pd, Ag, and Cs) in fuel particles (kernels, buffer, coating layers) and fuel elements (graphite matrix) by SEM/microprobe (WDX); look for evidence of fission product accumulations at the IPyC/SiC interface, fission product attack of SiC, and fission products outside the fuel particles. These are established techniques; however, the sensitivity of the device in a radiation environment may limit the detection level and accuracy. Mount preparation may smear or dissolve key fission products and make interpretation difficult if low concentrations are important.

Finally, the PIE will generate low-level waste, remote-handled low-level waste, and spent nuclear fuel. Radioactive liquid effluents will be generated as well. These items will be disposed of through the ORNL (or other institution's) waste stream.

PIE TASK-20: Waste Handling: Collect, package, and dispose of wastes and spent fuel generated during the conduct of the AGR PIE. This waste is to be handled in accordance with laboratory standards. Complete the required documentation. PIE reporting will be conducted through normal program channels, both formal (written reports) and informal (E-mail, Web sites, and teleconferences).

PIE TASK-21: Reporting: Disseminate the findings, results, and problems of the PIE task by both formal and informal reporting. Support the program requests for specific information, clarifications, and impact assessments. The deliverables for the PIE task will consist of equipment development plans, experimental test plans, activity or equipment documentation reports, and a PIE final report. The content of these plans and reports follows:

Equipment Development Plan. The equipment development plans detail the path to be followed for equipment design and fabrication, the required performance of the equipment, and the principal or theory of operation. They provide an estimate of the obtainable accuracy and measurement uncertainty, cost, and expected developmental time. They should allow the reader to determine if an apparatus has a reasonable chance of performing the intended task at an acceptable cost and schedule. These reports are limited to a single topic.

Experimental Test Plan. The experimental test plans detail the task to be accomplished, the item to be tested and its subcomponents, the required accuracy of the measurements, documentation requirements, cost, and schedule. In particular, a test plan will be used to guide the conduct of the PIE task, generally on a capsule-by-capsule basis.

Task or Equipment Documentation Report. The individual task or equipment reports document the work performed and the results of an activity. They serve as a record of the final result of a task or activity and allow the reader to reference the details of a piece of equipment or understand the process of a particular activity. They are generally short and limited to a narrow topic.

Quick-Look Reports. The quick-look reports are interim reports that document partial progress on an important task. The main purpose is to document the collected data and provide the current status of an investigation so that program decisions on overall task direction can be made. The material in these reports generally ends up as a chapter(s) in the final report.

PIE Final Report. The PIE final report details the work performed under the experimental test plan, documents the results and data, and provides a historical record for use by other program participants. It is a comprehensive document that allows the reader to understand what was done, how it was done, and the results obtained.

Table C.3 lists the likely deliverables for the tasks that have been identified in this Appendix. Figures C.2 through C.9 show tentative schedules for the AGR-1 through AGR-8 PIEs.

Appendix D. DETAILED FUEL PERFORMANCE MODELING TASK LISTING

A key element of the AGR Fuel Development and Qualification Program Plan is the development of fuel performance models. Many groups have attempted to model the performance of coated-particle fuels. These efforts have not resulted in a comprehensive model capable of predicting fuel performance with sufficient accuracy to directly facilitate fuel design or replaced 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 more first principles based fuel performance models of coated-particle fuel (either with UO_2 or UCO) that can be used to
  - guide current and future particle designs,
  - assist in irradiation and safety experiment planning,
  - predict observed fuel failures, and
  - 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; and
- benchmark these models/codes against U.S. and international irradiation and safety experiments where possible.

As part of our work, we assume that we will leverage other DOE and international efforts where possible (e.g., NERI, INERI, AFCI, IAEA CRP, and HTR-TN). In addition, we assume that verification and validation (V&V) of the codes directly will be part of this program. Finally, we also assume that irradiation space will be available in other capsules from the fuel and materials irradiation and/or the fission product transport working groups to irradiate special material specimens and fueled compacts for model improvement and ultimate V&V.

The development of fuel performance models requires a fundamental understanding of potential failure mechanisms and how these mechanisms depend on the irradiation conditions and the materials comprising the fuel. Accurate fuel performance modeling will also require good materials properties and constitutive relations. Table D.1 summarizes the key fuel failure mechanisms associated with TRISO-coated-particle fuel and how these mechanisms depend on reactor service conditions and the particle design and performance parameters. The following failure mechanisms under irradiation were considered: (1) pressure vessel failure, (2) fast-neutron-induced cracking of IPyC, (3) IPyC partial debonding, (4) kernel migration, and (5) fission product attack. Under accident conditions, fission product attack, SiC thermal decomposition, an increase in SiC permeability/SiC degradation, oxidation of the SiC layer, and rapid energy deposition were considered. Table D.2 summarizes the important material properties that are 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. In addition, comments are made concerning the fact that existing NERI proposals will cover much of this information.

The scope of this section is limited to activities needed to support fuel performance modeling. However, as indicated in Table D.1, 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

Table D.1. 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 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	UO_2 only. Not important for UCO. Reasonably well understood
Diffusive release through intact layers	Temperature Burnup Temperature gradient Time at temperature	Chemical state/transport behavior of fission products Microstructure of SiC SIC thickness	Could be more important at high burnup in LEU fuels because of greater yield of noble metals (e.g., Ag) from plutonium fissions. More important during accident conditions
Fission product attack	Temperature Burnup Temperature gradient Time at temperature	Fission product transport behavior Diffusion? Buffer densification and cracking? 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

Table D.1. (continued)

Failure mechanism	Reactor service conditions	Particle design and performance parameters	Comments
Parameters that strongly influence the failure mechanism			
Corrosion of SiC by CO	Temperature Burnup Time at temperature	Kernel type (UO_2, UCO) IPyC performance	CO is generated in particles with UO_2 kernels. At elevated temperatures, CO can attack the SiC layer if the IPyC layer is porous or has failed.
SiC thermal decomposition	Temperature Time at temperature	SiC thickness Microstructure of SiC	Not important in traditional accident envelope (peak temperature <1600°C)
Increase in SiC permeability/SiC degradation	Burnup Temperature Fluence	Microstructure of SiC? 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 heatup
Oxidation of SiC layer	Partial pressure of oxygen Temperature Time at temperature	Thickness of SiC layer Microstructure of layer	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 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

Note: ? indicates a potential parameter.

Table D.2. Key material properties needed for fuel performance modeling

Property	Current state of knowledge	Importance in modeling	How to measure	Other comments
Irradiation performance				
PyC anisotropy	Known to be critical to characterize PyC behavior. Ability to measure it accurately and precisely is in question	All key properties are thought to depend on anisotropy	X-ray, laser Raman, optical (questioned), other NDE techniques (acoustic?, UV,IR? magnetic?)	Critical QC/QA measure of acceptable PyC; needs development
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	Specific NERI projects may provide these data
PyC irradiation-induced creep	Uncertain with a factor of 5 uncertainty 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)	Specific NERI projects may provide these data
Poisson's ratio in creep	Reasonably well known. Literature data range from 0.3 to 0.5. Best estimate is 0.4. Probably is a function of density. Unclear if it is a function of anisotropy.	Has modest effect on stress in PyC layer	Special specimens	Specific NERI projects may provide these data
Strength of PyC	Data vary significantly. Some data 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	Specific NERI projects may provide these data

Table D.2. (continued)

Property	Current state of knowledge	Importance in modeling	How to measure	Other comments
Irradiation performance				
Strength of SiC	Data vary significantly. Need data as a function of density, neutron fluence, irradiation temperature, and microstructure (large grain vs small grain and columnar vs equiaxed). Microstructure is a function of deposition conditions. Data are available for Chinese SiC. German data suggest that irradiation can reduce strength. The United States has correlated a lot of data and concludes that 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 use axial compression of a cylindrical plug inside SiC cylindrical sample. Key issue is linkage of data to microstructure.	Specific NERI projects may provide these data
Interfacial bond strength between SiC and PyC	Very little is known. Historic value of ~50 MPa is used in calculations. Tends to agree reasonably well with values from SiC/SiC composites.	Critical to understanding the nature of debonding of the layers. The nature of the bond depends on the n of the fabrication process.	Special specimens and special punch/shear test to get bond strength.	Specific NERI projects may provide these data
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 (1000 to 1300°C) would be useful.	Lower importance given uncertainty in other parameters	Density (density gradient column) measurements	Specific NERI projects may provide these data
Irradiation-induced SiC creep	Limited data at low fluence.	Unknown given poor state of knowledge.	Split ring or bend strength relaxation techniques	Specific NERI projects may provide these data

Table D.2. (continued)

Property	Current state of knowledge	Importance in modeling	How to measure	Other comments
Irradiation performance				
Fission gas release from the kernel	Data on gas release is reasonably well known for UO_2. Little to no data on UC, especially at high burnup O.	Direct contributor to pressure in particle	Can be measured by crushing particles or online from "intentionally failed" particles	Would need loose irradiated 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	Would need loose irradiated UO_2 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 or irradiated fuel	
Accident performance: long-term heating/air ingress/rapid reactivity transients				
Thermal expansion coefficient of PyC	CTE is different in the two orientations in PyC and depends on the anistropy 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 size of sample adds to overall difficulty in measurement and ultimate uncertainty
Elastic modulus of PyC	Modulus is a function of anisotropy, fluence, density and temperature. Little 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	Specific NERI projects may provide these data
Elastic modulus of SiC	Data from fusion program show a 10% drop at reactor-relevant temperatures and radiation doses. Little data above 1000°C.	Critical for potential reactivity events where large temperature gradients may develop with the fuel particle	Resonant ultrasound spectroscopy or nanoindentation	Specific NERI projects may provide these data
Thermal expansion coefficient of SiC	Limited data suggest expansion is constant between 900 and 1300°C. No systematic dependence on coating temperature or neutron irradiation. The presence of free carbon in SiC can reduce CTE by 40%	Critical for potential reactivity events where large temperature gradients may develop with the fuel particle	Conventional techniques	Small size of sample adds to overall difficulty in measurement and ultimate uncertainty

Table D.2. (continued)

Property	Current state of knowledge	Importance in modeling	How to measure	Other comments
Accident performance: long-term heating/air ingress/rapid reactivity transients				
Fission product interactions with layers and potential degradation of properties	Unknown influence at present	Unknown at present	TBD	MIT will examine this influence under French INERI on particle fuel modeling
Buffer survivability	Failure of the buffer appears to be important to whether fission products get to the IPyC/SiC interface	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	This effect needs to be studied with the performance model before a definitive direction on the need for this work can be made
Kernel swelling under rapid energy deposition	Little data available under rapid energy deposition conditions	Kernel swelling and kernel-coating mechanical interaction may be critical to predicting failure in rapid reactivity transients	Part of PIE following reactivity transient testing	Unclear whether reactivity safety testing will be in scope of DOE AGR program

Note: ? indicates potential measurement technique.

working group. The research and development (R&D) needs for fuel performance and fission product transport will be combined as appropriate in this plan.

Description of R&D Needs, Associated Cost and Schedule, and Interfaces

3.4.1 Measurement of Anisotropy of PyC

Reliable, accurate, and precise methods are needed to characterize the anisotropy of PyC following fabrication and following irradiation. Existing data suggest that many of the key PyC material properties are a function of the anisotropy. Better methods, compared to the historical X-ray and optical techniques, are needed both for quality assurance (QA)/quality control (QC) and for fuel performance modeling.

Samples of PyC need to be fabricated and the anisotropy measured by both historical techniques and any new techniques Both disks and coated inert particles are recommended for study. The samples also need to be irradiated and their anisotropy measured to determine if there is an influence of irradiation on anisotropy. Piggyback locations in the AGR-3 capsule are available to provide for the irradiation of these samples.

Schedule and Cost:

Task	Schedule (months)
3.4.1.1 Fabrication	6
3.4.1.2 Anisotropy measurements (unirradiated)	6
3.4.1.3 Irradiation	24
3.4.1.4 Anisotropy measurements (irradiated)	9
3.4.1.5 Document results	6

Task	Cost ($K)
Fabrication	250
Irradiation	Covered in AGR-3 cost
Material property measurements (pre- and post-)	1,000
Documentation	100
Total	1,350

Interfaces: Interfaces to the fuel manufacture group are needed to define the types of PyC samples that need to be fabricated. Interfaces to the irradiations group are needed to accommodate the samples in piggyback irradiations to be the most cost-effective for this work. Interface to the fuel manufacture group is required to evaluate new anisotropy measurement techniques and to define requirements from the fuel fabrication perspective.

3.4.2 Thermomechanical and Thermophysical Properties of Coating Layers under Normal Operation

The thermomechanical and thermophysical properties of PyC and SiC listed in Table D.2 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 material 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 covered under a NERI project at a cost of ~$1M. Modest resources are required to interface with this work for its 3-year duration.

Task	Cost ($K)
3.4.2 Interface with NERI project	300

3.4.3 Thermochemical Properties of Kernel under Normal Operation

The thermochemical properties of the kernel listed in Table D.2 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 VHTR. In addition, CO release from UO_2 is needed at burnups in excess of 10% fissions per initial metal atom (FIMA) at relevant pebble bed modular reactor (PBMR) temperatures (up to 1300°C). Finally, measurements of kernel swelling for both UO_2 and UCO are needed, especially at high burnup.

Schedule and Cost

Task	Schedule (months)
3.4.3.1 Fabrication of UO_2 TRISO-coated particles	6
3.4.3.2 Irradiation of loose particles	24
3.4.3.3 Perform CO and kernel swelling measurements	18
3.4.3.4 Compile results	6

Task	Cost ($K)
Fabrication	200
Irradiation	
CO and kernel swelling measurement	1,750
Report results	150
Total	2,100

Note: This assumes no piggybacking in either fabrication or irradiation. If so, the cost would be much less. Also we assume here that we must construct the CO measurement experiment, although such a facility may still exist either in Europe or the United States.

Interfaces: Interfaces to the fuel manufacture group are needed to define the UO_2 coated particles that need to be fabricated. Interfaces to the irradiations group are needed to determine if a separate irradiation or piggyback irradiations are most cost-effective for this work. Interfaces with the PIE group are needed to define the actual measurement techniques and facilities.

3.4.4 Thermomechanical and Thermophysical Properties of Coating Layers under Accident Conditions

Table D.2 lists the properties needed to model the mechanical behavior of the coated particle under accident conditions. The thermal expansion coefficient and elastic modulus of PyC are needed as functions of fast fluence and temperature (1200–1800°C). Also needed are the corresponding properties of SiC. These are covered under the funded NERI project discussed in Task 3.4.2. Modest resources are required to track this work in the NERI.

In addition, buffer survivability under irradiation appears to be important in determining whether fission products get to the IPyC/SiC interface. Some data exist for modeling dimensional change of the buffer layer. More study of the effect with a fuel performance model is needed to scope out the critical material properties that are needed.

Task	Schedule (months)
3.4.4.1 Analytic study of buffer survivability	4

Task	Cost ($K)
Track NERI	225
Analytic study of buffer survivability	75
Total	300

Interfaces: Interfaces to the fuel manufacture group are needed to define the sample that needs to be fabricated. Interfaces to the irradiations group are needed to determine if a separate irradiation or piggyback irradiations are most cost-effective for this work. Interface to postirradiation examination (PIE) group is needed to review proposed measurement techniques/methods.

3.4.5 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 on the attack on the degradation of the properties of the layer has not been studied. Also, the kinetics of the reaction mechanism have not been definitely defined. Simple models assume that the particle is failed when ~ 50% of the layer has been attacked. This work is currently funded under INERI at a cost of ~$325 K.

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, 6–8 × 10^{25} neutrons/m^2). A technique to characterize this enhanced permeability is needed to accurately capture the effect in models. This would include better characterization of micro-structural changes in SiC following irradiation. *(Note that this work is still very preliminary and would benefit from additional review.)* Samples will be provided by the fuel manufacture group and irradiated as piggyback samples in the FPT-1 capsule.

Task 3.4.5.2a Characterize permeability and microstructures of SiC-coated inert particles irradiated to different fluences using classic permeability techniques (e.g., BET, mercury intrusion, CH_3I intrusion, and others).

Task 3.4.5.2b Characterize cesium permeability of irradiated SiC wafers—bulk interaction of cesium with irradiated SiC or sputter-coat cesium onto irradiated SiC—under accident temperatures.

Schedule and Cost:

Task		Schedule (months)
3.4.5.1	Follow work on Ag/SiC and Pd/SiC in French INERI	24
3.4.5.2.1	Fabrication of SiC particles and wafers	6
3.4.5.2.2	Microstructure and permeability measurements (unirradiated)	6
3.4.5.2.3	Irradiation	24
3.4.5.2.4	Microstructure and permeability measurements	9
3.4.5.2.5	Cesium permeability measurements	18
3.4.5.2.6	Document results	6

Task	Cost ($K)
Fabrication	100
Irradiation	Included in AGR-3
Permeability and microstructure characterization	250
Cesium permeability measurement	400
Documentation	100
Total	850

Kernel swelling data are needed under rapid energy deposition conditions. Kernel swelling and kernel coating mechanical interaction may be critical to predicting failure in reactivity transients. These data can be obtained as part of PIE following reactivity testing. It is unclear whether reactivity testing should be part of the DOE AGR program [currently not costed in the AGR program, but such testing would be very expensive (multimillion dollars)].

Interfaces: Interfaces to the fuel manufacture group are needed to define the samples that need to be fabricated. Interfaces to the PIE group are needed to formulate in more detail the permeability measurements. Interfaces are needed with the program manager to determine if reactivity testing is part of the AGR program.

3.4.6 Code Benchmarking and Improvement

There has been significant activity around the world (e.g., GA, INEEL, MIT, France) to develop improved fuel performance codes under normal operating conditions. Thus, we assume that the codes for normal operation are probably adequate, given the activity to update the component models for the various failure mechanisms. However, our preliminary assessment is that new accident condition codes are needed.

Benchmarking of fuel performance codes is needed. Pretest predictions and posttest 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 rerun to understand the influence of the improved data on the predicted behavior.

Schedule and Cost:

Task	Schedule (months)
3.4.6.1 Code benchmarking and modeling improvement (normal)	Level of effort
3.4.6.2 Requirements for accident performance codes	6
3.4.6.3 Develop new/improve accident performance codes	18
3.4.6.4 Historical benchmarking of improved accident performance	6
3.4.6.5 Report results of accident condition code	6
3.4.6.6 Benchmarking and modeling improvement (accidents)	Level of effort
Report results	Annually

Task	Cost ($K)
Code benchmarking/modeling improvement (assume 6 years @150/year)	900
Requirements for accident performance codes	100
Develop new/improve accident performance codes	600
Historical benchmarking of improved accident performance	500
Report results of accident condition code	100
Benchmarking and modeling improvement (accidents) (assume 6 years @ 150/year)	900
Total	3,100

Interfaces: Interfaces to the fuel manufacture group, irradiations group, and PIE group to provide data needed for benchmarking calculations. Model development activities have no direct interfaces per se.

3.4.7 Code V&V

The V&V of fuel performance codes is required for licensing. We will need an independent integral validation of the fuel performance codes. This will require an additional irradiation for code validation. The irradiation would cycle the fuel thermally and be designed so that some measurable level of fuel failure would occur against which the model could be validated. For example, conditions would be established that would provide measurable corrosion of the SiC by fission products or pressure vessel failure. Because of the statistical issues associated with fuel failure, a large sample size would be needed (~50,000 particles) in each irradiation. This work must be performed to support V&V, final reactor design, and ultimate licensing. (To do this validation scope with proof test fuel would be too late to support final design and licensing activities.)

For safety work, we propose to use a select number of compacts from the validation irradiation in traditional furnace heating tests to act as a data set for V&V of the accident fuel performance model.

The situation for beyond design basis accident testing and modeling is still open and will require determination of whether such work is part of the AGR program.

Schedule and Cost:

	Task	Schedule (months)
3.4.7.1	Fuel performance normal model verification	15
3.4.7.2	Fuel performance model validation irradiation	24
3.4.7.3	Model validation using data from irradiation capsule	9
3.4.7.4	Report results	6
3.4.7.4	Fuel performance accident model verification	15
3.4.7.5	Fuel performance accident validation heating tests	12
3.4.7.6	Fuel performance accident model validation	12
3.4.7.7	Report results	6

	Cost ($K)
Fuel performance normal model verification	400
Fuel performance validation irradiation	Covered in AGR-7 cost
Fuel performance normal model validation	300
Report results	150
Fuel performance accident model verification	600
Fuel performance accident validation testing—covered in safety test costs	
Fuel performance accident model validation	400
Report results	150
Total	2,000

(Note on schedule: verification activities can be performed off critical path. Validation activities must be tied to completion of specific tests; $4M in cost for irradiation and safety tests must be reflected in those sections.)

Interfaces: Interfaces to the fuel manufacture group, irradiations group, and PIE group are needed to provide data needed for validation calculations. Model verification activities have nodirect interfaces per se.

Appendix E. FISSION PRODUCT TRANSPORT AND SOURCE TERM DETAILS

This appendix provides a detailed description of R&D Task 3.5 and is organized accordingly.

R&D TASK 3.5 FISSION PRODUCT TRANSPORT AND SOURCE TERM

The goal of the Fission Product Transport and Source Term Working Group is to develop a research and development (R&D) plan that, when the work is successfully completed, will produce a technical basis for source terms under normal and accident conditions for the Very High Temperature Reactor (VHTR). The technical basis will be codified in design methods (computer models) validated by experimental data. The approach is to take credit for all fission product release barriers (i.e., kernels, coatings, graphite, primary coolant pressure boundary, and reactor building) in order 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 10^{-7} for normal operation and about 10^{-6} for core heatup accidents. Such stringent limits on particle failure are impractical given that the best that 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 low-enriched uranium (LEU) UO_2 at 10% fissions per initial metal atom (FIMA).[1]

The goals, assumptions, and requirements identified by the Fission Product Transport and Source Term Working Group are listed below.

Goals

- Provide a technical basis for the source terms under normal and accident conditions
- Bound the uncertainty in the source term
- 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 design data needs

Assumptions

- Advanced gas reactors (AGRs) will be designed such that the radionuclides are essentially retained in the core during normal operation and all credible accidents.
- Laboratory-scale testing is sufficient; full- or partial-scale prototypes are not required.
- AGRs will be designed and licensed to operate with a VLPC building.
- Some degree of credit must be taken for each of the principal fission product release barriers (i.e., kernels, coatings, matrix/graphite, primary coolant pressure boundary, containment building) to achieve a viable fuel and plant design.*
- TRISO fuel that meets fuel product specifications and in-core fuel performance requirements adopted for the VHTR can be provided.
- The accumulation of condensable radionuclides, especially silver and cesium isotopes, in the primary coolant circuits of direct-cycle high-temperature gas-cooled reactors (HTGRs), has significant design, operations and maintenance (O&M), and safety implications.
- Radiologically significant reactivity transients (i.e., those capable of compromising fuel integrity) are precluded by design; consequently, fuel performance and fission product release under these conditions need not be characterized experimentally.

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*Exclusive reliance on fission product retention by the particle coatings would necessitate impractical limits on coating failure: $\sim 10^{-7}$ during normal operation and $\sim 10^{-6}$ during core heatup accidents.

- The potential for large water ingress into the primary coolant circuit of a direct-cycle HTGR is much reduced compared a steam-cycle HTGR, but the potential for moderate water ingress (e.g., ~10,000 ppm H_2O) remains; consequently, the impact of water ingress on fuel performance and fission product transport will need to be characterized but at a reduced priority.
- The potential for air ingress into the primary coolant circuit of the VHTR exists so that 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 code validation will be required to assure reliable plant design and licensing. Fuel performance models, which are closely linked with fission transport models, are treated in Appendix D.
- Model development and code validation will be supported by the Department of Energy (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 design data needs 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 in-pile loop tests may be needed to obtain definitive silver and cesium plateout data on turbine and recuperator alloys under prototypical direct-cycle HTGR conditions and to generate representative samples for decontamination studies (the leading candidate for such tests is the PG-1 loop at NIIAR, Dimitrovgrad, R.F.).

Requirements

- Confirm the source terms under normal and accident conditions 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 Nuclear Regulatory Commission (NRC). Fuel performance models, which are closely linked with fission product transport models, are treated in Appendix D.
- Generate a comprehensive set of fission product transport data needs that defines the additional experimental data needed to confirm the source terms and validate the design methods.

The rationale for, and a brief description of, the recommended tasks are provided in the program elements below, along with deliverables, costs, and schedules.

3.5.1 Fission Gas Release, Including Iodine and Tellurium, from Failed Fuel Particles

The current database under normal operating conditions consists primarily of Test, Research, Isotopes General Atomics (TRIGA) measurements on laser-failed particles, including UCO kernels. Isothermal in-pile hydrolysis tests on LEU UCO fuel have been performed at Oak Ridge National Laboratory (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 high-flux reactor (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 required to develop and refine gas release models with uncertainties <4X at 95% confidence. No new experimental work on hydrolyzed fuel is recommended at this time, given the lower priority of water ingress in a direct-cycle AGR.

R&D Task 3.5.1.1: Measure fission gas release (Kr, Xe, I, and Te) from exposed LEU/natural UCO kernels [compacts seeded with "designed-to-fail" (DTF) particles, which are reference UCO kernels with a 10- to 15-μm PyC seal coat] irradiated under near normal HTGR flux over a range of temperatures. Update appropriate models in the SURVEY code.

Deliverable: Report results of measurements of fission gas release from exposed kernels and provide model improvements in the SURVEY code. Prepare final report on HFR B1 gas release.

Sample Materials and Irradiation Conditions Required: A total of 24 special compacts with reference driver fuel are needed. All compacts will contain reference LEU UCO/NUCO particles to serve as driver fuel. Of these, 16 compacts will be seeded with LEU UCO DTF particles, and 8 compacts will be seeded with NUCO DTF particles to a level of 10^{-3} to 10^{-2} failed fraction per compact. It is requested that the fabrication campaign for the DTF compacts produce a supply of compacts for use in future irradiations such as the in-pile loop testing in Task 3.5.10.1, where 12–20 DTF compacts may be required, depending on the reactor to be employed. It is suggested that a supply of approximately 60 DTF compacts and 800 unbonded DTF particles (600 LEU UCO and 200 NUCO) be produced.

Four compacts will be irradiated in each of six individually swept cells in one advanced test reactor (ATR) capsule: two cells at 1250°C (one with LEU UCO DTF compacts and one with NUCO DTF compacts), two cells at 1100°C (one LEU UCO DTF and one NUCO DTF), one cell at 900°C (LEU UCO DTF), and one cell at 700°C (LEU UCO DTF). The LEU UCO DTF compacts should be irradiated to 22% FIMA and 4.5×10^{25} neutrons/m^2, and the NUCO DTF compacts should be irradiated to 3% FIMA and 4.5×10^{25} neutrons/m^2. It is required that the effluent gas lines be filtered, probably with charcoal, to trap metal fission products and that the entire effluent line, up and including the filter, be recoverable for hot cell examination. With four compacts per cell, there is ample room for piggyback samples. These irradiations will address Tasks 3.5.1.1, 3.5.1.2, 3.5.2.1, 3.5.2.2, and 3.5.3.1, as well as PyC and SiC samples in support of Fuel Performance Modeling Tasks.

Schedule and Cost:

Task	Schedule (months)	Cost ($K)
3.5.1.1.1 Fabrication	2	75
3.5.1.1.2 Irradiation/online gas release	32	2,275
3.5.1.1.3 Measure iodine/tellurium release	6	616
3.5.1.1.4 Final report on HFR B1 gas release	3	75
3.5.1.1.5 Report results/upgrade models	4	100
3.5.1.1.6 Plan/monitor/evaluate		150

Total cost: $3,291K

Note that the irradiation and PIE-related costs described in this appendix are captured in the Sect. 4.1 cost roll-up tables under the Fuel and Materials Irradiation, and Safety Testing and PIE program elements. This approach was taken because the irradiation and PIE work costs are essentially service costs (the cost estimates originate with the service provider) and to minimize the potential for double counting of costs.

Interfaces:

Task	Interfaces
3.5.1.1.1 Fabrication	Fuel manufacture—type and number of samples to be fabricated and schedule and cost of fabrication
3.5.1.1.2 Irradiation/online gas release	Irradiation—type of irradiation capsule, online gas release measurements, temperature measurements, and schedule and cost of irradiation
3.5.1.1.3 Measure iodine/tellurium release	Safety Testing and PIE—methods and accuracies of measurements of iodine and tellurium release, and schedule and cost of measurements
3.5.1.1.4 HFR B1 gas release report	Analysis of Ben Myers' reports (see Task 3.5.2.1.4)
3.5.1.1.5 Report results	Safety Testing and PIE—data analysis and reporting
3.5.1.1.6 Task planning, monitoring, evaluation	Irradiation/PIE—test and measurement specification documents and data evaluation

R&D Task 3.5.1.2: Measure fission gas, including iodine, from irradiated, 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 code.

Deliverable: Report results of fission gas release, including iodine, from irradiated fuel under dry CCCD conditions.

Schedule and Cost:

Task	Schedule (months)	Cost ($K)
3.5.1.2.1 Fabrication	NA	NA
3.5.1.2.2 Irradiation/online gas release	NA	NA
3.5.1.2.3 Reactivate test specimens	2	105
3.5.1.2.4 Measure gas and iodine release	9	700
3.5.1.2.5 Report results/upgrade models	4	100
3.5.1.2.6 Plan/monitor/evaluate		150

Total cost: $1,055K

Schedule and costs for sample material fabrication and irradiation are included in Task 3.5.1.1.

Interfaces:

Task	Interfaces
3.5.1.2.1 Fabrication	Fuel manufacture—type and number of samples to be fabricated and schedule and cost of fabrication
3.5.1.2.2 Irradiation/online gas release	Irradiation—type of irradiation capsule, online gas release measurements, and schedule and cost of irradiation
3.5.1.2.3 Reactivate test specimens	Safety Testing and PIE—facility, schedule, and cost
3.5.1.2.4 Measure gas and iodine release	Safety Testing and PIE—laser puncturing particles, methods and accuracy of measurements of fission gas and iodine release under CCCD conditions, and schedule and cost of measurements
3.5.1.2.5 Report results/upgrade models	Irradiation/Safety Testing and PIE—data analysis and reporting
3.5.1.2.6 Task planning, monitoring, evaluation	Irradiation/PIE—test and measurement specification documents and data evaluation

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 test capsules. There are German data for Cs, Sr, and Ag in UO_2 kernels of intact particles that were irradiated under near real-time conditions as well as limited laboratory data on cesium release from ThO_2 kernels. Data on metal release from 20% FIMA LEU UCO kernels may possibly be derived from HFR B1/Cell 1. Gaps in the data needed 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, which is used to calculate full-core metal fission product release under normal operating conditions; the TRAMP/COPAR code, which is used to calculate "hot spot" metal fission product transport under normal operating conditions (and used for capsule analysis); and the SORS code, which is 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 <10X at 95% confidence.

R&D Task 3.5.2.1: Measure and model fission metal release from LEU/natural UCO fuel kernels in failed and intact particles under near real-time irradiation. The DTF particles irradiated in piggyback samples and in seeded fuel compacts will be used. The principal information on metal release will be from mass balances derived from radiochemical measurements of fission products transported to the irradiation capsule interior. Data from the postirradiation examination (PIE) of HFR B1/Cell 1 will be analyzed to derive fission metal transport properties. Update the fission metal release correlations in the TRAFIC/COPAR and TRAMP/COPAR codes.

Deliverable: Report results of measurements of effective fission metal diffusivities in UCO kernels under normal irradiation conditions and document modeling improvements.

Sample Materials and Irradiation Conditions Required: The same 24 compacts seeded with DTF particles to a level of 10^{-3} to 10^{-2} failed fraction per compact identified in Task 3.5.1.1 will be used and irradiated as described in Task 5.3.1.1. In addition, about 100 DTF unbonded particles (50 LEU UCO and 50 NUCO) will be irradiated in piggyback samples to 22% FIMA (LEU UCO), and 3% FIMA (NUCO) and 4.5×10^{25} neutrons/m^2. The piggyback sealed capsules will be placed in the fuel bodies irradiated at 1100°C.

Schedule and Cost:

Task	Schedule (months)	Cost ($K)
3.5.2.1.1 Fabrication	NA	NA
3.5.2.1.2 Irradiation	NA	NA
3.5.2.1.3 Measure fission metal effective diffusivities	6	600
3.5.2.1.4 Analyze data from HFR B1/Cell 1	3	60
3.5.2.1.5 Report results/upgrade models	4	100
3.5.2.1.6 Plan/monitor/evaluate		150

Total cost: $910K

The costs of sample fabrication and irradiation are included in Task 3.5.1.1.

Interfaces:

Task	Interfaces
3.5.2.1.1 Fabrication	Fuel manufacture—types and numbers of samples to be fabricated, and schedule and cost of fabrication
3.5.2.1.2 Irradiation	Irradiation—types of irradiation capsules, temperature measurement, and schedule and cost of irradiation
3.5.2.1.3 Measure effective diffusivities	Safety Testing and PIE—methods and accuracies of measurements, schedule, and cost
3.5.2.1.4 Analyze data from HFR B1/Cell 1	Final summary report based on Ben Myers' status reports and conference papers (Task 1.1.4)
3.5.2.1.5 Report results/ upgrade models	Safety Testing and PIE—data analysis and reporting
3.5.2.1.6 Task planning, monitoring, evaluation	Irradiation/PIE—test measurement specification documents and data evaluation

R&D Task 3.5.2.2: Measure and model fission metal release from LEU/natural UCO fuel kernels in failed and intact particles irradiated 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 code.

Deliverable: Report results of measurements of effective fission metal diffusivities in UCO kernels under dry CCCD conditions and document modeling improvements.

Schedule and Cost:

Task	Schedule (months)	Cost ($K)
3.5.2.2.1 Fabrication	NA	NA
3.5.2.2.2 Irradiation	NA	NA
3.5.2.2.3 Measure fission metal effective diffusivities	See Task 3.5.1.2.4	700
3.5.2.2.4 Report results/upgrade models	4	100
3.5.2.2.5 Plan/monitor/evaluate		150

Total cost: $950K

Schedule and cost for sample fabrication and irradiation are included in Task 3.5.1.1.

Interfaces:

Task	Interfaces
3.5.2.2.1 Fabrication	Fuel manufacture—types and numbers of samples to be fabricated, and schedule and cost of fabrication
3.5.2.2.2 Irradiation	Irradiation—types of irradiation capsules, temperature measurement, and schedule and cost of irradiation
3.5.2.2.3 Measure effective diffusivities	Safety Testing and PIE—methods and accuracies of measurements, and schedule and cost
3.5.2.2.4 Report results/upgrade models	Safety Testing and PIE—data analysis and reporting
3.5.2.2.5 Task planning, monitoring, evaluation	Irradiation/PIE—test and measurement specification documents and data evaluation

3.5.3 Fission Product Effective Diffusivities in Particle Coatings

The fuel particle coatings, particularly the SiC coating, are the primary barriers to the release of 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 pyrocarbon 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 are dependent on the structures of the SiC and PyC coatings, which vary with the coating manufacturing process. Therefore, fission product diffusivities must 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, fluence, and as-manufactured coating attributes to reduce uncertainties to <10X at 95% confidence.

R&D Task 3.5.3.1: Measure diffusivities of Ag, Cs, and Sr in SiC and I, Te, Xe, and Kr in PyC as a function of temperature, fluence and as-manufactured coating attributes. Unbonded, irradiated particles and particles deconsolidated from irradiated fuel elements with low particle failure rates will be heated and diffusivities determined by measuring time signatures of fission product releases from the particles. Diffusivities of fission gases in PyC will be measured at very high temperatures where SiC is thermally degraded.

Deliverable: Report results of measurements of fission product diffusivities in particle coatings.

Sample Materials and Irradiation Conditions Required: About 100 unbonded LEU UCO particles will be irradiated in piggyback samples at about 1100°C, 22% FIMA, and 4.5×10^{25} neutrons/m^2 in the same irradiation as Task 3.5.1.1. Fuel compacts irradiated to 22% FIMA and 4.5×10^{25} neutrons/m^2 from fuel demonstration/qualification tests with low leach/burn/leach measurements of particle failure will also be used as sample material.

Schedule and Cost:

Task	Schedule (months)	Cost ($K)
3.5.3.1.1 Fabrication	NA	NA
3.5.3.1.2 Irradiation	NA	NA
3.5.3.1.3 Measure fission product diffusivities	12	1,058
3.5.3.1.4 Report results	3	75
3.5.3.1.5 Plan/monitor/evaluate		150

Total cost: $1,283K

It has been assumed here that irradiated fuel for this task will be available from irradiations that have already been planned (e.g., Task 3.5.1.1 and fuel performance/qualification testing) so that no new costs for fuel fabrication and irradiation have been generated.

Interfaces:

Task	Interfaces
3.5.3.1.1 Fabrication	Fuel manufacture—as-fabricated fuel characteristics/quality
3.5.3.1.2 Irradiation	Irradiation—irradiation conditions/online fission gas release
3.5.3.1.3 Reactivate	Safety Testing and PIE—facility, schedule, and cost
3.5.3.1.4 Measure fission product diffusivities	Safety Testing and PIE—methods and accuracies of measurements of fission product diffusivities data analysis and reporting
3.5.3.1.5 Report results	Safety Testing and PIE—data analysis and reporting
3.5.3.1.6 Task planning, monitoring, evaluation	Irradiation/PIE—test and measurement specification documents and data evaluation

R&D Task 3.5.3.2: Update correlations for fission product diffusivities in coatings in the TRAFIC/COPAR, TRAMP/COPAR, and SORS codes.

Deliverable: Description of updated models for fission product diffusivities in the TRAFIC/COPAR, TRAMP/COPAR, and SORS codes.

Schedule and Cost:

Task	Schedule (months)	Cost ($K)
3.5.3.2.1 Update models	1	25
3.5.3.2.2 Test models	2	50
3.5.3.2.3 Report results	1	25

Total cost: $100K

Interfaces:

Task	Interfaces
3.5.3.2.1 Update models	Fuel Performance Modeling—interface between fuel performance and fission product transport models
3.5.3.2.2 Test models	Fuel Performance Modeling—same as above
3.5.3.2.3 Report results	Fuel Performance Modeling—same as above

3.5.4 Fission Product Diffusivities/Sorptivities in Graphite

Fuel element graphite can significantly attenuate the release of fission metals and preclude the release of actinides from the core during normal operation and during CCCD transients. The present correlations for fission metal diffusivities in core graphite are derived largely from laboratory measurements on unirradiated nuclear graphites and from 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 data are limited for cesium and strontium on irradiated graphite and irradiated compact matrix material. The available data indicate that 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. Gaps in the needed data are mainly in the area of irradiated graphite. In addition, large uncertainties exist in the correlations of fission metal transport in graphite because many of the apparent variables are not treated explicitly. Sufficient single-effects test data are needed to develop and refine diffusivity and sorptivity correlations with uncertainties <10X at 95% confidence.

R&D Task 3.5.4.1: Measure diffusivities and sorptivities of Cs, Sr, Ag, and Pu in fuel-compact matrix and fuel element graphites as a function of temperature, fast fluence, and, as appropriate, coolant impurities, system pressure (for Ag), and the extent of graphite oxidation under normal operating and CCCD conditions.

Deliverable: Report of results of measurements of diffusivities and sorptivities of Cs, Sr, Ag, and Pu in irradiated compact matrix and core graphites.

Sample Materials and Irradiation Conditions Required: Samples will consist of a designed-to-fail LEU UCO compact fuel source surrounded by three concentric annuli: first, an annulus of compact matrix material, then an annulus of fuel element graphite (replacement for H-451) material, and finally an annular char-impregnated graphite sink. Six samples, each in a swept cell at a controlled temperature will

be irradiated in one ATR capsule. It is requested that one cell be irradiated at 1100°C to a fluence of 4.5×10^{25} neutrons/m^2, two cells be irradiated at 950°C (one to a fluence of 4.5×10^{25} neutrons/m^2 and one to a lower fluence), one cell at 800°C and 4.5×10^{25} neutrons/m^2, and two cells at 700°C (one at 4.5×10^{25} neutrons/m^2 and one at a lower fluence).

Schedule and Cost:

Task	Schedule (months)	Cost ($K)
3.5.4.1.1 Fabrication	4	50
3.5.4.1.2 Irradiation	32	2,275
3.5.4.1.3 Measure diffusivities, sorptivities	12	2,030
3.5.4.1.4 Report results	3	75
3.5.4.1.5 Plan/monitor/evaluate		300

Total cost: $4,730K

Interfaces:

Task	Interfaces
3.5.4.1.1 Fabrication	Fuel manufacture—type and number of samples to be fabricated, and schedule and cost of fabrication
3.5.4.1.2 Irradiation	Irradiation—type of irradiation capsule, sharing irradiation capsule with other materials testing, temperature measurements, and schedule and cost of irradiation
3.5.4.1.3 Measure diffusivities, sorptivities	Safety Testing and PIE—methods and accuracy of measurements, schedule and cost of measurements, and data analysis and reporting
3.5.4.1.4 Report results	Safety Testing and PIE—data analysis and reporting
3.5.4.1.5 Task planning, monitoring, evaluation	Irradiation/PIE—test and measurement specification documents and data evaluation

R&D Task 3.5.4.2: Improve model for fission metal transport in graphite in the TRAFIC/COPAR, TRAMP/COPAR, and SORS codes.

Deliverable: Report documenting model improvements and results of test calculations.

Schedule and Cost:

Task	Schedule (months)	Cost ($K)
3.5.4.2.1 Improve models	3	75
3.5.4.2.2 Test models	3	75
3.5.4.2.3 Report results	1	25

Total cost: $175K

Interfaces:

Task	Interfaces
3.5.4.2.1 Improve models	Fuel Performance Modeling—interface between fuel performance and fission product transport models
3.5.4.2.2 Test models	Fuel Performance Modeling—same as above
3.5.4.2.3 Report results	Fuel Performance Modeling—same as above

3.5.5 Fission Gas Release Validation Data

The SURVEY and SORS codes require data on fission gas release for code validation that are independent of the data used to develop the codes. The validation should assure that the predictive methods are accurate to within 4X 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 irradiation capsules. The noble gas release from FSV at the end-of-life was overpredicted by about a factor of 2, where hydrolysis may have been less severe than in lab tests. The noble gas release from Peach Bottom Core 2 at the end-of-life was underpredicted by a factor or 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 >10X higher as-manufactured coating defects than required for modern direct-cycle HTGRs. Fission gas release from irradiation capsules 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 from data used to develop the SURVEY and SORS codes.

R&D Task 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 under near normal HTGR flux over a range of temperatures for validation of the SURVEY code.

Deliverable: Report of the measurement of fission gas release under normal operating conditions.

Sample Materials and Irradiation Conditions Required: A total of 36 fuel compacts, containing both LEU UCO and NUCO particles seeded with both missing buffer LEU UCO and NUCO particles (most representative particle failures in reference fuel) at a level of 10^{-3} to 10^{-2} are needed. Six compacts will be irradiated in each of six individually swept and temperature-controlled cells in one ATR capsule. Four cells will be irradiated to 22% FIMA and 4.5×10^{25} neutrons/m^2, at the following average temperatures: one at 1250°C, one at 1100°C, one at 900°C, and one at 700°C. In addition, it is requested to irradiate two compacts, one at either end of the capsule, to reduced burnups, one at 1100°C and one at 700°C. It is also requested to perform periodic thermal cycling, to the maximum possible (as much as 100°C, perhaps), by varying the sweep gas composition to approximate representative time-temperature histories in the reactor core. It is required that the effluent gas lines be filtered, probably with charcoal, to trap metal fission products and that the entire effluent line, up and including the filter, be recoverable for hot cell examination. These irradiations will also be used to address Tasks 3.5.5.3, 3.5.6.1, and 3.5.6.3.

Schedule and Cost:

Task	Schedule (months)	Cost ($K)
3.5.5.1.1 Fabrication	2	50
3.5.5.1.2 Irradiation/online gas release	32	1,975
3.5.5.1.3 Measure iodine/tellurium release	6	610
3.5.5.1.4 Report results	3	75
3.5.5.1.5 Plan/monitor/evaluate		150

Total cost: $2,866K

The in-pile loop tests called for in program elements 3.5.9 and 3.5.10 would also contribute important gas and metallic release validation data, especially at high pressure and high mass flow.

Interfaces:

Task	Interfaces
3.5.5.1.1 Fabrication	Fuel manufacture—type and number of samples to be fabricated, and schedule and cost of fabrication
3.5.5.1.2 Irradiation/on/line gas release	Irradiation—type of irradiation capsule, online gas release measurements, temperature measurements, and schedule and cost of irradiation
3.5.5.1.3 Measure iodine/ tellurium release	Safety Testing and PIE—methods and accuracies of measurements of iodine and tellurium release, and schedule and cost of measurements
3.5.5.1.4 Report results	Irradiation/Safety Testing and PIE—data analysis and reporting
3.5.5.1.5 Task planning, monitoring, evaluation	Irradiation/PIE—test and measurement specification documents and data evaluation

R&D Task 3.5.5.2: Validate the SURVEY code for calculating fission gas release by performing pretest predictions and posttest calculations with the code and comparing the results with measurements.

Deliverable: Report on the validation of the SURVEY code.

Schedule and Cost:

Task	Schedule (months)	Cost ($K)
3.5.5.2.1 Pretest calculations	4	100
3.5.5.2.2 Posttest calculations	4	100
3.5.5.2.3 Compare calculations/measurements and report results	2	50

Total cost: $250K

Interfaces:

Task	Deliverable
3.5.5.2.1 Pretest calculations	Fuel manufacture—fuel characteristics and as-manufactured fuel quality Irradiation—irradiation conditions planned, including in-pile loop tests in Task 3.5.10
3.5.5.2.2 Posttest calculations	Irradiation—irradiation conditions as run, including in-pile loop tests in Task 3.5.10
3.5.5.2.3 Compare calculations/ measurements and report results	Irradiation/Safety Testing and PIE—release results

R&D Task 3.5.5.3: Measure fission gas, including iodine, from irradiated reference fuel with known failure fraction under CCCD conditions for validation of the SORS code. The atmosphere for the CCCD conditions will be determined as the core design and accident analyses evolve.

Deliverable: Report of the measurement of fission gas release under accident conditions.

Schedule and Cost:

Task	Schedule (months)	Cost ($K)
3.5.5.3.1 Fabrication	NA	NA
3.5.5.3.2 Irradiation/online gas release	NA	NA
3.5.5.3.3 Reactivate test specimens	2	105
3.5.5.3.4 Measure fission gas release, including iodine	9	562
3.5.5.3.5 Report results	2	50
3.5.5.3.6 Plan/monitor/evaluate		150

Total cost: $867K

The costs of fabrication and irradiation of fuel sample materials used in this task have been included in Task 3.5.5.1.

Interfaces:

Task	Interfaces
3.5.5.3.1 Fabrication	Fuel manufacture—type and number of samples to be fabricated, and schedule and cost of fabrication
3.5.5.3.2 Irradiation/online gas release	Irradiation—type of irradiation capsule, online gas release measurements, temperature measurements, and schedule and cost of irradiation
3.5.5.3.3 Reactivate test specimens	Safety Testing and PIE—facility, schedule, and cost
3.5.5.3.4 Measure fission gas release	Safety Testing and PIE—methods and accuracy of measurements of fission gas release, including iodine, under dry CCCD conditions, and schedule and cost of measurements
3.5.5.3.5 Report results	Safety Testing and PIE—data analysis and reporting

R&D Task 3.5.5.4: Validate the SORS code for fission gas release by performing pretest predictions and posttest calculations with the code and comparing the results with the measurements.

Deliverable: Report of the validation of the SORS code.

Schedule and Cost:

Task	Schedule (months)	Cost ($K)
3.5.5.4.1 Pretest calculations	2	50
3.5.5.4.2 Posttest calculations	2	50
3.5.5.4.3 Compare calculations/measurements and report results	1	25

Total cost: $125K

Interfaces:

Task	Interface
3.5.5.4.1 Pretest calculations	Fuel manufacture—fuel characteristics and quality
3.5.5.4.2 Posttest calculations	Safety Testing and PIE—CCCD conditions planned Safety Testing and PIE—CCCD conditions as run
3.5.5.4.3 Compare calculations/measurements	Safety Testing and PIE—data analysis and reporting results

3.5.6 Fission metal release validation data

The TRAFIC/COPAR and TRAMP/COPAR codes for calculating fission metal release under normal operating conditions and the SORS code 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 assure that the predictive methods are accurate to within 10X at 95% confidence. The validity of the codes for predicting fission metal release under normal operating conditions have been assessed by applying them to predict the observed metal release in operating HTGRs (Peach Bottom Core 2 and FSV) and in irradiation capsules and in-pile loops. Most of the available data are for cesium, with a small amount of silver and strontium data. In general, the releases of fission metals were

underpredicted 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 assessed systematically.

R&D Task 3.5.6.1: Measure fission metal release from irradiation capsules containing LEU/natural UCO fuel with known failure under normal operating conditions for validation of the TRAFIC/COPAR and TRAMP/COPAR codes.

Deliverable: Report on the measurement of fission metal release under normal irradiation conditions.

Sample Materials and Irradiation Conditions Required: The same five compacts with missing buffer particles fabricated and irradiated in Task 3.5.5.1 will be used in this task.

Schedule and Cost:

Task	Schedule (months)	Cost ($K)
3.5.6.1.1 Fabrication	See Task 3.5.5.1.1	NA
3.5.6.1.2 Irradiation	See Task 3.5.5.1.2	NA
3.5.6.1.3 Measure fission metal release	6	600
3.5.6.1.4 Report results	3	75
3.5.6.1.5 Plan/monitor/evaluate		150

Total cost: $825K

This assumes one multicell capsule in ATR is to be shared by Tasks 3.5.5.1 and 3.5.6.1. Fuel fabrication and irradiation costs are shown under Task 3.5.5.1.

The in-pile loop tests called for in Tasks 3.5.9 and 3.5.10 would also contribute important gas and metallic release validation data, especially data at high pressure and high mass flow.

Interfaces:

Task	Interfaces
3.5.6.1.1 Fabrication	Fuel manufacture—types and numbers of samples to be fabricated, and schedule and cost of fabrication
3.5.6.1.2 Irradiation	Irradiation—types of irradiation capsules, temperature measurement, and schedule and cost of irradiation
3.5.6.1.3 Measure fission metal release	Safety Testing and PIE—methods and accuracy of measurements, and schedule and cost
3.5.6.1.4 Report results	Safety Testing and PIE—data analysis and reporting
3.5.6.1.5 Task planning, monitoring, evaluation	Irradiation/PIE—test and measurement specification documents and data evaluation

R&D Task 3.5.6.2: Validate the TRAFIC/COPAR and TRAMP/COPAR codes for fission metal release by performing pretest predictions and posttest calculations with the code and comparing the results with the measurements.

Deliverable: Report of validation of the TRAFIC/COPAR and TRAMP/COPAR codes.

Schedule and Cost:

Task	Schedule (months)	Cost ($K)
3.5.6.2.1 Pretest calculations	4	100
3.5.6.2.2 Posttest calculations	4	100
3.5.6.2.3 Compare calculations/measurements and report results	2	50

Total cost: $250K

Interfaces:

Task	Interfaces
3.5.6.2.1 Pretest calculations	Fuel manufacture—fuel characteristics, and quality Irradiation—irradiation conditions planned, including in-pile loop tests in Task 3.5.10
3.5.6.2.2 Posttest calculations	Irradiation—irradiation conditions as run, including in-pile loop tests in Task 3.5.10
3.5.6.2.3 Compare calculations/ measurements and report results	Irradiation/Safety Testing and PIE—release results

R&D Task 3.5.6.3: Measure fission metal release from irradiated reference fuel heated under CCCD conditions for validation of the SORS code. The atmosphere for the CCCD conditions will be determined as the core design and accident analyses evolve.

Deliverable: Report of the measurement of fission metal release under accident conditions.

Schedule and Cost:

Task	Schedule (months)	Cost ($K)
3.5.6.3.1 Fabrication	NA	NA
3.5.6.3.2 Irradiation/online gas release	NA	NA
3.5.6.3.3 Measure fission metal release	See Task 3.5.5.3.4	562
3.5.6.3.4 Report results	2	50
3.5.6.3.5 Plan/monitor/evaluate		150

Total cost: $762K

The costs of fabrication and irradiation of fuel sample materials for this task have been included in Task 3.5.5.1. Cost for accident testing measurements are shared between Task 3.5.5.3.4 and Task 3.5.6.3.3.

Interfaces:

Task	Interfaces
3.5.6.3.1 Fabrication	Fuel manufacture—same as Task 3.5.6.1.1
3.5.6.3.2 Irradiation/online gas release	Irradiation—same as Task 3.5.6.1.1
3.5.6.3.3 Measure fission metal release	Safety Testing and PIE—methods and accuracy of measurements of fission metal release under dry CCCD conditions, and schedule and cost of measurements
3.5.6.3.4 Report results	Safety Testing and PIE—data analysis and reporting
3.5.6.3.5 Task planning, monitoring, evaluation	Irradiation/PIE—test and measurement specification documents and data evaluation

R&D Task 3.5.6.4: Validate the SORS code for fission metal release by performing pretest predictions and posttest calculations with the code and comparing the results with the measurements.

Deliverable: Report of the validation of the SORS code.

Schedule and Cost:

Task	Schedule (months)	Cost ($K)
3.5.6.4.1 Pretest calculations	2	50
3.5.6.4.2 Posttest calculations	2	50
3.5.6.4.3 Compare calculations/measurements and report results	1	25

Total cost: $125K

Interfaces:

Task	Interfaces
3.5.6.4.1 Pretest calculations	Fuel manufacture—fuel characteristics, quality Safety Testing and PIE—CCCD conditions planned
3.5.6.4.2 Posttest calculations	Safety Testing and PIE—CCCD conditions as run
3.5.6.4.3 Compare calculations/ measurements	Safety Testing and PIE—data analysis and reporting results

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 O&M, especially for a direct-cycle plant. Correlations currently available that describe the deposition behavior of condensable radionuclides on structural metals have very large uncertainties (>>10X). 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 (tungsten). The current database is inadequate to estimate the potential importance of diffusion of deposited radionuclides into the interior of structural metals (indiffusion) at operating temperatures. Data are needed to characterize the deposition of Cs, Ag, I, and Te on structural metals. Correlations are needed which 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 <10X at 95% confidence.

R&D Task 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.

Deliverable: Report results of measurements of the deposition characteristics of Cs, Ag, I, and Te on structural metals.

Sample Materials and Irradiation Conditions Required: These out-of-pile loop tests will not require any fuel samples or irradiation services.

Schedule and Cost:

Task	Schedule (months)	Cost ($K)
3.5.7.1.1 Develop experimental technique, using radioactive tracers	12	300
3.5.7.1.2 Measure deposition characteristics	36	2,625
3.5.7.1.3 Report results	3	75

Total cost: $3,000K

The experimental costs are split between deposition (Task 3.7.7.1) and reentrainment (Task 3.5.8.1).

Interfaces:

Task	Interfaces
3.5.7.1.1 Develop experimental technique	Safety Testing and PIE—explore methods and accuracy of measurements
3.5.7.1.2 Measure deposition characteristics	Safety Testing and PIE—make measurements
3.5.7.1.3 Report results	Safety Testing and PIE—data analysis and reporting

R&D Task 3.5.7.2: Update correlations for fission product deposition on structural metals in the PADLOC code.

Deliverable: Report documenting model improvements and results of test calculations.

Schedule and Cost:

Task	Schedule (months)	Cost ($K)
3.5.7.2.1 Improve models	2	50
3.5.7.2.2 Test models	2	50
3.5.7.2.3 Report results	1	25

Total cost: $125K

Interfaces:

Task	Interfaces
3.5.7.2.1 Improve models	Fuel Performance Modeling—informal contact
3.5.7.2.2 Test models	Fuel Performance Modeling—informal contact
3.5.7.2.3 Report results	Fuel Performance Modeling—informal contact

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 (>>10X). The liftoff database was obtained in blowdown tests wherein the test specimens were mechanically removed from the loop or reactor in which the plateout activity was originally deposited. These ex situ blowdown 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 blowdown to that during normal operation—and, to a lesser extent, the duration of the blowdown. No correlation between the fractional liftoff and the blowdown temperature or the humidity of the helium was evident. Ex situ liftoff data from the Japan Atomic Energy Research Institute (JAERI) (small contaminated samples cut from a component of the OGL-1 loop) suggest that reentrainment may be relatively modest even for very large shear ratios, SR > 100. High-quality liftoff data were obtained in COMEDIE BD-1; however, the materials of construction and service conditions were for the steam-cycle modular high-temperature gas-cooled reactor (MHTGR), and the effect of dust was minimized by use of a full-flow filter (a planned second test with dust was not performed). The extent to which plateout activity may be removed during rapid depressurization transients must be quantified for VHTR materials of construction and service conditions. Correlations are needed for I, Sr, Cs, Te, and Ag as a function of SR, wall shear stress, blowdown 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 <10X at 95% confidence.

R&D Task 3.5.8.1: Measure reentrainment of the key radionuclides just identified under the conditions specified in a series of out-of-pile loop tests.

Deliverable: Report of the results of measurements of reentrainment.

Sample Materials and Irradiation Conditions Required: These out-of-pile loop tests will not require any fuel samples or irradiation services.

Schedule and Cost:

Task	Schedule (months)	Cost ($K)
3.5.8.1.1 Develop experimental technique using radioactive tracers	12	300
3.5.8.1.2 Measure reentrainment	36	2,625
3.5.8.1.3 Report results	3	75

Total cost: $3,000K

The experimental costs are split between deposition (Task 3.5.7.1) and reentrainment (Task 3.5.8.1).

Interfaces:

Task	Interfaces
3.5.8.1.1 Develop experimental technique	Safety Testing and PIE—explore methods
3.5.8.1.2 Measure liftoff	Safety Testing and PIE—make measurements
3.5.8.1.3 Report results	Safety Testing and PIE—data analysis and reporting

R&D Task 3.5.8.2: Improve reentrainment model in POLO code.

Deliverable: Report reentrainment model improvements in the POLO code.

Schedule and Cost:

Task	Schedule (months)	Cost ($K)
3.5.8.2.1 Improve models	2	50
3.5.8.2.2 Test models	2	50
3.5.8.2.3 Report results	1	25

Total cost: $125K

Interfaces:

Task	Interfaces
3.5.8.2.1 Improve models	Fuel Performance Modeling—informal contact
3.5.8.2.2 Test models	Fuel Performance Modeling—informal contact
3.5.8.2.3 Report results	Fuel Performance Modeling—informal contact

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 (<10X at 95% confidence) for normal operating conditions and for CCCD transients. The data must be independent of those used to develop 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°C–500°C, well below the surface temperatures in the gas turbine. 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 these data have been 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 have not been analyzed. 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 turbine and recuperator materials of construction and need to be performed under VHTR service conditions; it is highly desirable to include a small, simulated turbine because there are no existing plateout data on rotating machinery. It is also desirable to include the effects of dust on the plateout distribution.

R&D Task 3.5.9.1: Measure radionuclide plateout from integral tests under normal operating and CCCD conditions in an in-pile loop (consider the PG-1 loop at NIIAR). The effects of dust should be quantified. Under normal operating conditions, the radionuclides in order of decreasing importance are Ag, Cs, I, and Te. The primary coolant range should be 100°C to 850°C, the helium coolant pressure should be >1 MPa, and the partial pressure of the radionuclides should be <<10 pPa. Under CCCD conditions, the radionuclides of interest in decreasing order of importance are I, Cs, Ag, and Te. The primary coolant temperature range should be 100°C to 700°C; the primary coolant pressure should be ≥0.1 MPa; the primary coolant should contain He, CO, N_2, H_2, and H_2O; and the partial pressure of the radionuclides should be ~1 nPa. Two tests are planned: one under "clean" conditions and the other with dust added. The costs are shared with Task 3.5.10.1, which measures liftoff under core conduction conditions following the deposition phase.

Deliverable: Report of the plateout results of integral tests in an in-pile loop conducted to provide an independent data set for the validation of the plateout model in the PADLOC code.

Sample Materials and Irradiation Conditions Required: Driver fuel and a source of fission products (compacts containing DTF particles) are required. The quantities of these materials required will depend of the size of the reactor, but will likely be approximately 3 side-by-side fuel stacks of 8 compacts, each containing two DTF compacts (1×10^{-3} to 1×10^{-2} failure level), for a total of 6 DTF compacts and 18 driver fuel compacts per test.

Schedule and Cost:

Task		Schedule (months)	Cost ($K)
3.5.9.1.1	Specify test program and select loop for plateout and liftoff tests (2 tests)	12	75
3.5.9.1.2	Conduct tests	36	1,650
3.5.9.1.3	Analyze test data and report results	12	150
3.5.9.1.4	Task planning, monitoring, evaluation		225

Total cost: $2,100K

Interfaces:

Task	Interfaces
3.5.9.1.1 Specify test program and select loop for plateout and liftoff tests (2 tests)	Safety Testing and PIE—test planning
3.5.9.1.2 Conduct tests	Safety Testing and PIE—methods and accuracy of measurements, cost and schedule Fuel manufacture—specification of fuel to be fabricated
3.5.9.1.3 Analyze test data and report results	Fuel Performance Modeling—analysis of test data
3.5.9.4 Test planning, monitoring, evaluation	Irradiation/PIE—test and measurement specification documents and data evaluation

R&D Task 3.5.9.2: Validate the PADLOC code by performing pretest predictions and posttest calculations and comparing results with measurements. Analyze data from the COMEDIE SR-1 test, and compare code calculations with data.

Deliverable: Report results of validation of the PADLOC code and analysis of the COMEDIE SR-1 test.

Schedule and Cost:

Task	Schedule (months)	Cost ($K)
3.5.9.2.1 Pretest calculations (release, plateout)	3	37.5
3.5.9.2.2 Posttest calculations	12	150
3.5.9.2.3 Compare calculations/measurements and report results	3	37.5
3.5.9.2.4 Analyze data from COMEDIE SR-1 and compare code calculations with data	6	100

Total cost: $325K

Interfaces:

Task	Interfaces
3.5.9.2.1 Pretest calculations	Fuel Performance Modeling—informal contact
3.5.9.2.2 Posttest calculations	Fuel Performance Modeling—informal contact
3.5.9.2.3 Compare calculations and measurements	Fuel Performance Modeling—informal contact
Analyze COMEDIE SR-1	Fuel Performance Modeling—informal contact

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 assure predictive accuracy within 10X at 95% confidence. The data must be independent of those used to develop 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 blowdown test of the CPL 2/4 in-pile loop, <0.5% liftoff of the deposited activity was

observed; however, the maximum SR was only 1.08. Moreover, the CPL 2/4 loop contained an inordinate amount of metal oxide aerosol, so the data are likely to be biased high. Considerable additional liftoff data were generated by the PIE of the COMEDIE BD-1 test, wherein 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 made at Peach Bottom, Dragon, and AVR. 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 Idaho National Engineering and Environmental Laboratory (INEL), 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. There are also German data from measurements made in the AVR; however, the considerable dust in the AVR resulted primarily from mechanical attrition of the circulating fuel spheres, and the applicability of these data to a prismatic VHTR is debatable. Limited semiquantitative data are also available from the General Atomics (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 (>10X) the amount of liftoff observed in ex situ blowdown 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 assure that dust effects do not preclude validating design methods to predict fission product transport within the primary coolant circuit to within an accuracy of 10X at 95% confidence.

R&D Task 3.5.10.1: Measure radionuclide reentrainment (liftoff) from integral tests during rapid depressurization transients in an in-pile loop (consider the PG-1 loop at NIIAR). Radionuclides in order of decreasing importance are I, Sr, Cs, Te, and Ag. The effects of dust should be quantified. The primary coolant temperature range should be 100°C to 850°C, SRs in the range 0.5 to 5, and blowdown durations 1 to 10 min. Two tests are planned: one under "clean" conditions and the other with dust added. The cost is shared with Task 3.5.9.1.

Deliverable: Report of the liftoff results of integral tests in an in-pile loop conducted to provide an independent data set for the validation of the reentrainment model in the POLO code.

Sample Materials and Irradiation Conditions Required: Common to Task 3.5.9.1.

Schedule and Cost:

Task		Schedule (months)	Cost ($K)
3.5.10.1.1	Specify test program and select loop for plateout and liftoff tests (2 tests)	12	75
3.5.10.1.2	Conduct tests	36	1,650
3.5.10.1.3	Analyze test data and report results	12	150
3.5.10.1.4	Plan/monitor/evaluate		225

Total cost: $2,100K

Interfaces:

Task	Interfaces
3.5.10.1.1 Specify test program and select loop for plateout and liftoff tests (2 tests)	Safety Testing and PIE—test planning
3.5.10.1.2 Conduct tests	Safety Testing and PIE—methods and accuracy of measurements, and cost and schedule Fuel manufacture—specification of fuel to be manufactured
3.5.10.1.3 Analyze test data and report results	Fuel Performance Modeling—formal contact
3.5.10.1.4 Task planning, monitoring, evaluation	Irradiation/PIE—test and measurement specification documents and data evaluation

R&D Task 3.5.10.2: Validate the POLO codes by performing pretest predictions and posttest calculations with the codes and comparing the results with the measurements.

Deliverable: Report results of validation of POLO codes.

Schedule and Cost:

Task	Schedule (months)	Cost ($K)
3.5.10.2.1 Pretest calculations (release, plateout, liftoff)	3	37.5
3.5.10.2.2 Posttest calculations	12	150
3.5.10.2.3 Compare calculations/measurements and report results	3	37.5

Total cost: $225K

Interfaces:

Task	Interfaces
3.5.10.2.1 Pretest calculations	Fuel Performance Modeling—informal contact
3.5.10.2.2 Posttest calculations	Fuel Performance Modeling—informal contact
3.5.10.2.3 Compare calculations/measurement	Fuel Performance Modeling—informal contact

3.5.11 Fission Product Transport in a Vented Low-Pressure Containment

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 PAG dose limits at the EAB 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 made 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 light-water reactor (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). These LWR data may be applicable to the VLPC, but parameters, such as aerosol particle size and concentration, and fission product chemical forms will need to be evaluated. 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. The extent to which LWR data on radionuclide transport, especially transport within containment buildings, are applicable to the VLPC must be determined.

R&D Task 3.5.11.1: Determine by thermodynamic code calculations or small-scale experiments, if necessary, the chemical species to be expected under CCCD conditions in the VLPC. Investigate the applicability of methods and data for modeling radionuclide transport within LWR containments to conditions within the VLPC under CCCD conditions. Utilize, to the extent possible, LWR methods and data to calculate radionuclide transport and deposition within the VLPC under CCCD conditions. If necessary, supplement these methods with the required separate effects measurements at a relatively small scale. The radionuclides of interest are I, Cs, Te, Sr, and Ag.

Deliverable: Report of scoping studies on fission product transport and deposition within a VLPC under CCCD conditions.

Sample Materials and Irradiation Conditions Required: No fuel materials or irradiation services are required for this task.

Schedule and Cost:

Task	Schedule (months)	Cost ($K)
3.5.11.1.1 Determine fission product chemical forms in VLPC	3	75
3.5.11.1.2 Determine applicability of LWR analytical methods	2	50
3.5.11.1.3 Conduct, if necessary, small-scale, separate effects testing to supplement available analytic methods	6	400
3.5.11.1.4 Calculate fission product transport and deposition in VLPC under CCCD conditions	3	75
3.5.11.1.5 Report results	2	50

Total cost: $650K

Interfaces:

Task	Interfaces
3.5.11.1.1 Fission product chemical forms	Fuel Performance Modeling—thermodynamic calculations Safety Testing and PIE—experiments
3.5.11.1.2 Applicability of LWR methods	Fuel Performance Modeling—informal contact
3.5.11.1.3 Small-scale deposition experiments	Safety Testing and PIE—experiment design and conduct
3.5.11.1.4 Deposition calculations	Fuel Performance Modeling—informal contact
3.5.11.1.5 Report results	Fuel Performance Modeling—report review

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 (DF) 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 Task 3.5.12.1: Conduct laboratory tests of the pressure relief train filter to measure DF for key radionuclides. Select several candidate filter mediums and test over a range of bed depths for the expected range of blowdown stream conditions. The effects of temperature and coolant chemistry must be treated explicitly. The chemical composition of the key radionuclides (I, Sr, Cs, Te, and Ag) must also be determined. Revise as necessary the predictive model, and use it to optimize the relief valve train filter design.

Deliverable: Report of DF results for pressure relief train filter, revised predictive model, and optimized filter design.

Sample Materials and Irradiation Conditions Required: No fuel materials or irradiation services are required for this task.

Schedule and Cost:

Task	Schedule (months)	Cost ($K)
3.5.12.1.1 Conduct DF tests	6	300
3.5.12.1.2 Improve predictive model	2	50
3.5.12.1.3 Optimize filter design	2	50
3.5.12.1.4 Report results	2	50

Total cost: $450K

Interfaces:

Task	Interfaces
3.5.12.1.1 Conduct DF tests	Safety Testing and PIE—test design and conduct tests
3.5.12.1.2 Improve predictive model	Fuel Performance Modeling—informal contact
3.5.12.1.3 Optimize filter design	Safety Testing and PIE—filter design
3.5.12.1.4 Report results	Safety Testing and PIE—analyze and report results Fuel Performance Modeling—report review

3.5.13 International Cooperation

A number of AGR R&D programs are ongoing or in the process of being organized on an international basis. Cooperation with these international programs provides a potential for contributing to the resolution of a number of the R&D tasks identified above while saving program cost and schedule time.

R&D Task 3.5.13.1: Review international AGR R&D programs and facilities/capabilities and recommend opportunities for cooperation to reduce program costs and/or schedule in the resolution of fission product transport and source term R&D tasks.

Deliverable: Report on recommendations for international cooperation in the area of fission product transport and source term.

Sample Materials and Irradiation Conditions Required: No fuel materials or irradiation services are required for this task.

Schedule and Cost:

Task	Schedule (months)	Cost ($K)
3.5.13.1.1 Review international programs/facilities/capabilities	2	25
3.5.13.1.2 Make recommendations/report results	1	25

Total cost: $50K

Interfaces:

Task	Interfaces
3.5.13.1.1 Review programs	Fuel manufacture—contacts with international programs Irradiation—contacts with international programs Safety Testing and PIE—contacts with international programs Fuel Performance Modeling—contacts with international programs
3.5.13.1.2 Recommendations/report	All Working Groups—review report

REFERENCE

1. IAEA-TECDOC-978, *Fuel Performance and Fission Product Behavior in Gas-Cooled Reactors*, November 1997.

Appendix F. TOPICAL ISSUE DISCUSSIONS

This appendix provides four discussions on selected topical issues that were considered in the formulation of the plan. The first discussion provides supporting justification for focusing early fuel development efforts on UCO (uranium oxycarbide) fuel for prismatic modular reactors (PMRs), addressing the need for additional data on UCO fuel for PMRs and the relevance to pebble-bed reactors (PBRs) of existing international data and planned future work on the performance of low-enriched uranium (LEU) UO_2 TRISO fuel. The second discusses the demonstration of fuel performance margins and recommends that irradiation and testing of initial variants from the fuel production development effort not include consideration of margin demonstration. The third discusses accelerated irradiation and recommends that the acceleration factor not exceed a factor of 3 over conditions anticipated in plant service. The fourth briefly discusses a number of risks associated with commercial deployment of advanced gas reactor (AGR) technology.

F.1 FUEL DEVELOPMENT CONSIDERATIONS FOR PRISMATIC AND PEBBLE BED REACTOR CONCEPTS

The following sections briefly summarize the development status of fuel for PMR and PBR concepts that could be utilized in a very-high-temperature reactor (VHTR). The overall objective of the U.S. development program is to bring the state of the technology to the point where a manufacturer could confidently proceed to replicate an existing product, establish high-volume production capability, and qualify the resulting fuel for reactor service. In the absence of an existing VHTR design and corresponding fuel performance requirements, the initial fuel development will focus on achieving performance that can be expected to be reached with reasonable confidence based on existing data [e.g., maximum operating temperatures of 1250–1300°C, maximum burnup of 16–22% fissions per initial metal atom (FIMA), maximum fast fluence of $4–5 \times 10^{21}$ n/cm^2]. Additional development is required for both LEU UCO prismatic fuel for PMRs and LEU UO_2 fuel for PBRs. Work on the latter fuel is ongoing in other countries and available to the United States via international agreements. Thus, the U.S. program will focus on the development of LEU UCO fuel to meet PMR requirements. Assuming this development is successful, it is anticipated that the capabilities of PBR concepts would also be enhanced by the use of UCO fuel.

F.1.1 Development Status of Fuel for PMR Concepts

As noted in the following discussion, the reference fuel for prismatic reactor concepts in the United States is based on use of an LEU UCO TRISO fissile particle. This fuel form was selected in the early 1980s for large high-temperature gas-cooled reactor (HTGR) concepts using LEU, and the selection was reconfirmed for modular designs in the mid-1980s. Limited existing irradiation data on LEU UCO TRISO fuel indicate the need for a substantial improvement in performance with regard to in-pile gaseous fission product release. Existing accident testing data on LEU UCO TRISO fuel are extremely limited, but it is generally expected that performance would be similar to that of LEU UO_2 TRISO fuel if performance under irradiation were successfully improved. Thus, fuel development is needed before the technology is in a position where a manufacturer could confidently proceed to replicate an existing product for high-volume production. Experience with LEU UO_2 TRISO fuel provides confidence that the required development can be conducted expeditiously and successfully.

F.1.1.1 Background

A strong interest in PMR concepts remains evident among utilities and generating companies, particularly in the United States and Japan. Primary reasons for the continuing interest are expectations

of superior economics due to higher power ratings and simpler refueling systems relative to PBR concepts. Higher power ratings have resulted from the ability to locate reactivity control systems in fueled regions and to establish a more optimum annular core geometry. Simpler refueling systems are possible as a result of the absence of at-power fuel recirculation used in PBR concepts. Another important factor in this interest is existing technology and experience with PMR fuel, plant systems, and components in the United States because of prior work on PMRs. It is generally recognized that to obtain these advantages, PMR concepts place different and in some cases more demanding requirements on the fuel based on the following considerations:

- Batch (full core or core segment) refueling places a greater value on average spent fuel burnup to increase refueling intervals and meet plant capacity factor goals, resulting in a higher optimum average spent fuel burnup for PMRs.
- The ratio of peak to average discharge burnup is higher for a batch-refueled PMR, which removes a large fraction of the core at each refueling, relative to a multi-pass PBR, which removes individual fuel spheres from recirculation when the measured discharge burnup exceeds a specified value.
- The heat transfer characteristics of PMRs have typically led to higher temperature gradients and higher peak fuel temperatures relative to mixed-mean coolant outlet temperatures.

These considerations, in addition to the different fuel form (compacts in hexagonal blocks vs spheres) and differences in experience and technology base among countries pursuing HTGR technology, have led to significant differences in fuel performance requirements and fuel technology development directions between PMRs and PBRs. Initial U.S. HTGR fuel technology was based on carbide fuel forms. In the early 1980s, as HTGR technology was transitioning from high-enriched uranium (HEU) fuel to LEU fuel, trade studies were conducted and fuel design selection meetings were held within the general framework of the DOE HTGR program. An initial effort* focused on LEU prismatic designs for large HTGRs resulted in the selection of UCO kernels for the fissile particles and thorium oxide (ThO_2) for the fertile particles. The primary reason for selection of the UCO kernel over UO_2 was reduced CO pressure, allowing higher burnup for equivalent coating thicknesses and reduced potential for kernel migration, an important failure mechanism in earlier fuels. A subsequent assessment in the mid 1980s considering modular HTGR concepts again reached agreement on UCO for the fissile particle for a prismatic design. In the early 1990s, plant cost-reduction studies led to a decision to change the fertile material from thorium to natural uranium, primarily because of a lower long-term decay heat level for the natural uranium fissile particles. Ongoing economic optimization in combination with anticipated capabilities of the UCO particles resulted in peak fissile particle burnup projections of 26% FIMA in steam cycle and gas turbine concepts.

Japan is also pursuing a PMR concept, primarily directed toward the supply of industrial process heat. UO_2 fuel evolving from the German technology, restricted to very low burnup, has been selected. For example, the High-Temperature Test Reactor (HTTR) in Japan is limited to 3% FIMA for the first core, with ongoing fuel development expected to allow higher burnups in subsequent cores.

F.1.1.2 PMR service conditions

Currently, the most advanced PMR design, the Gas-Turbine Modular Helium Reactor (GT-MHR) concept under development for weapons plutonium disposition in Russia, with a design variant for commercial application using LEU fuel, can serve as a starting point for establishing PMR fuel service conditions. The information presented in Table F.1 is based on the most recent generally available data and should be considered for illustration purposes only. Reference service conditions for the fuel development program will be established in the course of producing the irradiation test specifications.

*GA-A17123, Selection of LEU/Th Reference Fuel for the HTGR-SC/C Lead Plant, May 1983.

Table F.1. GT-MHR service conditions[a] (reactor outlet temperature 850°C)

Parameter	
Maximum operating fuel temperature, °C	1320
Maximum fissile particle burnup, % FIMA	26
Maximum fast neutron fluence, neutrons/cm^2	5×10^{21}
Maximum accident fuel temperature, °C	1600

[a]General Atomics presentations at ANS Gas Reactor Technology Course, ANS Winter Meeting, November, 2002.

F.1.1.3 Existing irradiation data

The most comprehensive compilation of international data and experience with LEU coated-particle fuel is provided in a document* produced by an International Atomic Energy Agency (IAEA) Coordinated Research Project (CRP) on validation of predictive methods for fuel and fission product behavior. This CRP, conducted between 1991 and 1996, included participants from China, France, Germany, Japan, the United Kingdom, and the United States. An overview of the U.S. fuel irradiation program of relevance to LEU UCO fuel is provided on pages 100–106 of the IAEA document. In the late 1970s and early 1980s, several capsules containing multiple LEU fuel types (UCO, UC_2, UO_2, UO_2*) were irradiated in support of selecting a reference fuel. Early data from these irradiations, along with other considerations, resulted in the selection of UCO fuel for the fissile particle, as noted earlier. In the mid 1980s, several capsules containing UCO fuel, including a small fraction of intentionally failed and designed-to-fail particles, were irradiated to investigate the effect of moisture ingress on fission product release from failed particles. In the late 1980s, the first of an intended series of capsules focusing on LEU UCO fuel irradiation and containing sufficient quantities to qualify the fuel for service was irradiated. The performance of this capsule, in terms of in-pile gaseous fission product release and failed fuel fraction, was far below requirements for the plant design; and the postirradiation examination and testing scope was adjusted to focus on understanding the reasons for the failures. The planned additional capsules were cancelled, and fuel development activities were redirected toward identifying changes in the fuel to address the cause of the failures. An assessment of the U.S. LEU TRISO fuel experience in comparison with the German results† provides more detailed information on the U.S. irradiation experience.

In the early 1980s, Germany fabricated and irradiated LEU UCO compacts to burnups of up to 22% FIMA with good results [gaseous fission product release/birth ratios (R/B) generally less than 10^{-6}]. The results of these tests are seen as a basis for confidence that the gaseous fission product R/B will be low for LEU UCO fuel planned for production in the U.S. program. Germany also loaded more than 5000 spheres containing HEU UCO particles into the Arbeitsgemeinschraft Versuchreaktor (AVR) in the late 1970s. These spheres appear to have performed well, although there are no data on gas release specific to the UCO spheres because they were mixed in with a large number of other fuel types.

F.1.1.4 Accident condition testing

As noted earlier, several capsules of UCO containing a small fraction of initially failed and designed-to-fail particles were irradiated to study the effect of moisture on fission product release from failed fuel. These experiments provided data regarding response to water ingress events and could be of

*IAEA-TECDOC-978, *Fuel Performance and Fission Product Behavior in Gas-Cooled Reactors*, November 1997, available electronically at http://www.iaea.org/inis/aws/htgr/index.html.

†INEEL/EXT-02-00300, Key Differences in the Fabrication, Irradiation and Testing of U.S. and German TRISO-coated Particle Fuel and Their Implications of Fuel Performance, June 2002.

interest depending on the plant design being considered (e.g., for a steam cycle power plant or steam reformer for hydrogen production).

Data on high-temperature testing of irradiated fuel to address the response to primary coolant depressurization accidents are very limited for LEU UCO fuel. Small amounts of irradiated fuel have been heated, but the fuel from the LEU UCO capsule irradiated in the late 1980s was not subjected to heatup testing because of the large failure fraction experienced during irradiation. Thus heatup test results from LEU UO_2 pebble bed fuel discussed in the following section are generally referenced as a basis for expecting comparable high fission product retention by the LEU UCO prismatic fuel.

F.1.2 Development Status of Fuel for PBR Concepts

The modern PBR concept ties directly to the German fuel development program. The German program benefited considerably from early efforts in the United Kingdom and the United States. Data and experience developed in Russia, Japan, and China on LEU UO_2 TRISO fuel provide additional complementary support for the understanding of UO_2 fuel performance under normal operation and accident conditions and proof that high-quality UO_2 fuel can be fabricated at other facilities based on variations of the German design and process. The combined body of international data provides a demonstration of proof-of-principle for UO_2 coated-particle fuel and a basis for confidence that PBR fuel with performance at a level comparable to that of the modern German fuel can be manufactured.

F.1.2.1 Background

Coated-particle fuel for HTGRs has been under development for more than 40 years in many countries. The particle kernels studied have included HEU and LEU; thorium, uranium/thorium mixtures and plutonium; in oxide, carbide, and oxycarbide forms. Early coatings included a single high-density pyrocarbon layer, a buffer (low-density pyrocarbon) layer with high-density isotropic pyrocarbon outer layer (BISO), and other combinations. Most of the later development focused on the TRISO design, as noted earlier. Most of the countries with active HTGR programs in the 1980s and 1990s became focused on LEU UO_2 fuel very similar to typical PBR fuel in the major parameters, as illustrated in Table F.2.

Table F.2. Comparison of primary UO_2 reference coated-particle dimensions (μm)[1]

PBR/ country	Kernel diameter	Buffer thickness	IPyC thickness	SiC thickness	OPyC thickness	Fuel form
PBR	**500**	**95**	**40**	**35**	**40**	**Sphere**
South Africa	500	95	40	35	40	Sphere
Germany	500	95	40	35	40	Sphere
China	500	90	40	35	40	Sphere
Russia	500	95	75	60	60	Sphere
Japan	600	60	30	30	45	Block

As seen in Table F.2, the coated-particle designs being pursued in these countries are very similar to the PBR design, and thus fuel fabrication and testing data and experience are directly relevant to the PBR fuel. While the Russian reference design coating thicknesses differ considerably from those of the PBR design, a range of coating thicknesses that included the PBR values were used in the Russian test program. It should also be noted that Japan is planning to use a 500-μm kernel for the second core of HTTR. The testing data include a broad range of normal operation conditions. Additional large-quantity performance data have been generated through operation of reactors using UO_2 coated-particle fuel, including the AVR and THTR in Germany and the HTTR in Japan.

F.1.2.2 PBR service conditions

Currently, the most advanced PBR design concept, the PBMR, under development in South Africa, can be used as a starting point for establishing PBR fuel service conditions. The PBMR core design is continuing to evolve, and much of the design detail is proprietary. The information presented in Table F.3 is based on the most recent publicly available data, with interpretation regarding possible design limits; thus it should be considered for illustration purposes only.

Table F.3. PBMR service conditions (reactor outlet temperature 900°C)

Parameter	Nominal[a]	Design[b]
Maximum operating fuel temperature, °C	1182	1300
Maximum burnup, % FIMA	9	11
Maximum fast neutron fluence, neutrons/cm^2	2.16×10^{21}	2.5×10^{21}
Maximum accident fuel temperature, °C	1473	1600[c]

[a]E. J. Mulder, "PBMR: Opting for an annular reactor layout," IAEA 17th Meeting of International Working Group on Gas Cooled Reactors, November 2001.

[b]Estimated from nominal values and available nonproprietary data.

[c]PBMR, pty Doc. No. 010520, Rev. 2, "Pebble Bed Modular Reactor Nuclear Fuel (Nonproprietary Version)."

F.1.2.3 Existing irradiation data

Extensive irradiation programs of UO_2 coated fuel particles and fuel elements have been conducted in Germany, Japan, and Russia. Fuel produced in China is currently under irradiation in the Russian test reactor IVV-2M. The results of the irradiations in Germany, Japan, and Russia are reported in considerable detail in IAEA-TECDOC-978, with references to more detailed data. The early results of the irradiation of fuel from China are reported in the proceedings of a seminar held in China in March 2001.*

Figures F.1 through F.4 provide a simplified overview of the available data. The German testing followed broader developmental testing in the 1960s and 1970s and was primarily focused on supporting the design and licensing of the HTR-MODUL concept, covering both steam cycle and process heat options. Testing in China is in progress and is directed toward supporting the operation of the HTR-10 reactor. Testing in Russia was more exploratory and covered a wider range of conditions, including investigating the limits of the capability of the fuel. Testing in Japan was directed toward supporting the operation of the HTTR, primarily the first core loading.

As indicated in Fig. F.1, the irradiation temperatures span a broad range. The German data were intended to cover the range anticipated for the HTR-MODUL design, and the Chinese data are planned to support the HTR-10. As noted earlier, the Russian program was directed toward exploring the capability of the fuel and thus included higher temperatures and burnups. The program in Japan has been directed toward the use of HTGRs for high-temperature process heat and thus for higher coolant outlet temperatures.

As indicated in Fig. F.2, fuel specimens in Germany and Russia were taken to burnups in excess of 15%, considerably above the expected PBR average discharge burnup of 9% FIMA. The burnups for the fuel irradiations in Japan are considerably lower, consistent with the design conditions for the HTTR first core loading. The value given for China is the burnup achieved as of early 2001, with planned maximum burnup of approximately 11% FIMA at the completion of the test. The range of fast fluence exposure for the irradiations in Germany and Russia is shown in Fig. F.3 (data on fluence were not available for

*_Proceedings of the Seminar on HTGR Applications and Development, Beijing, China_, paper 21, available electronically at http://www.inet.tsinghua.edu.cn/english/HTR_meetings_S/CONTENTS.htm.

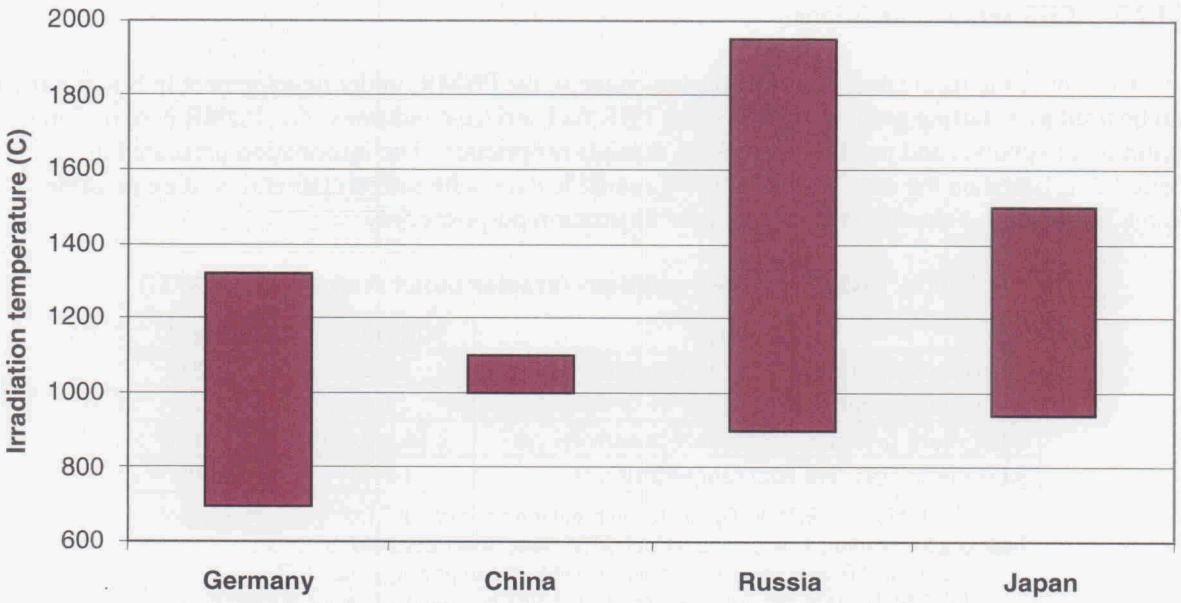

Fig. F.1. Irradiation temperature ranges.

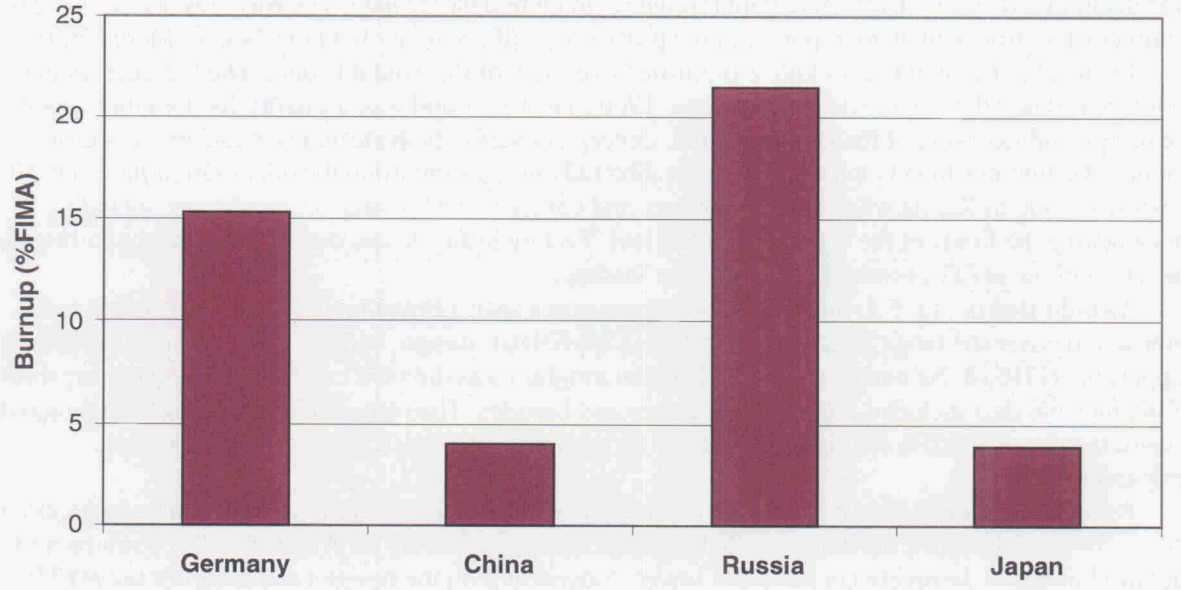

Fig. F.2. Burnup ranges.

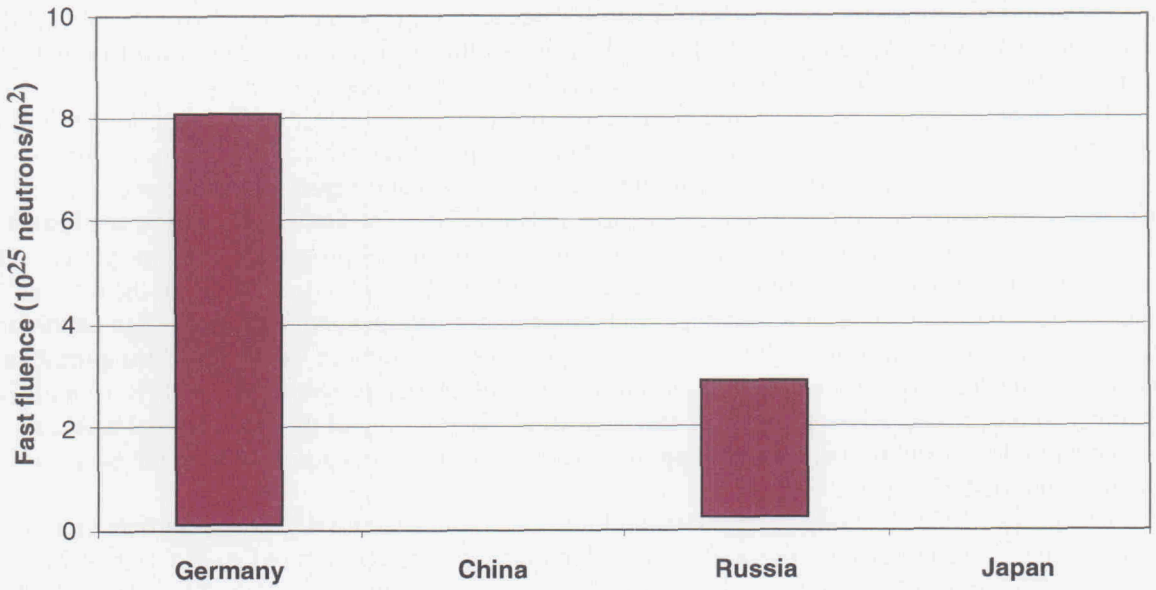

Fig. F.3. Fast fluence ranges.

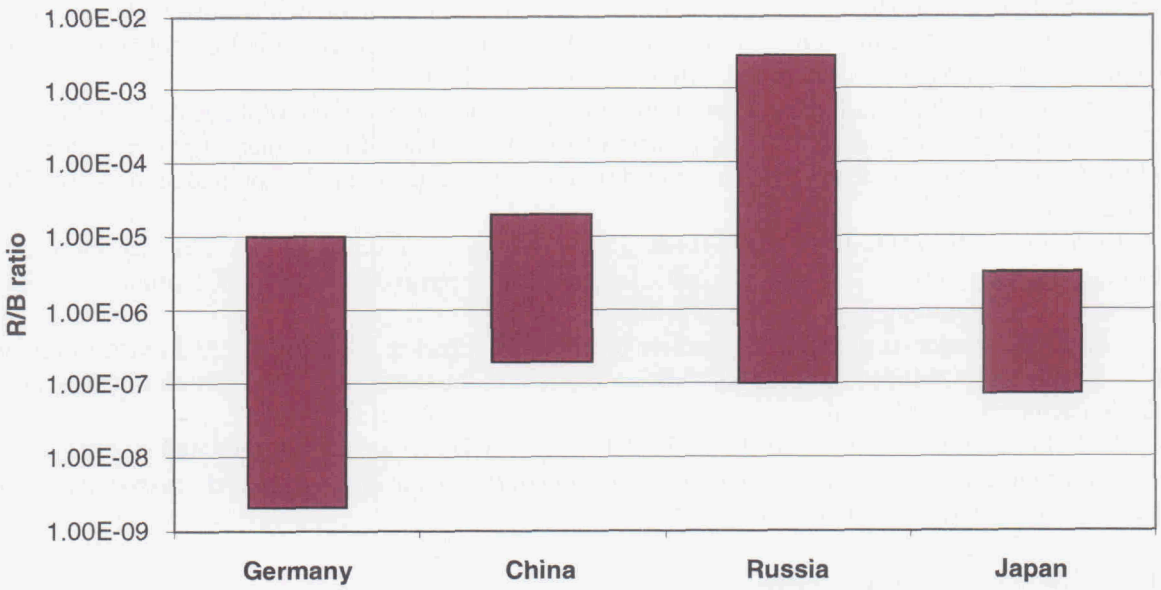

Fig. F.4. In-pile release-to-birth ratio.

the irradiations in China and Japan). The German fluence data are reported in terms of E > 16fJ, while the Russian data are reported for E > 32fJ; thus the Russian fluence levels would be considerably higher if reported for E > 16fJ, which is the basis for the PBR fast fluence specification.

The range of in-pile gaseous fission product R/B for the irradiation testing is shown in Fig. F.4. The data for Germany and China are reported as R/B of the isotope ^{85m}Kr, which typically runs slightly higher (<2X) than the ^{88}Kr R/B reported for the data from Russia and Japan. This difference is sufficiently small that the comparison is not significantly affected. The German data show the lowest in-pile release, consistent with the more mature status of the German fuel program in the time period of the testing. The much higher release of the Russian testing reflects the exploratory nature of the testing, which subjected the fuel to fluence, burnups, and temperatures well beyond expected design conditions. The data from Japan reflect the higher operating temperatures experienced by the fuel during irradiation. The data from China are for the early products of the fuel fabrication program for the HTR-10 initial fuel loading, and they reflect a lesser degree of maturity. Burn leach testing of the HTR-10 fuel batches showed an improvement in fuel quality of approximately an order of magnitude during the course of producing the first 25 batches.

Taken as a whole, the body of international fuel irradiation data on UO_2 coated-particle fuel of similar design, as summarized above and discussed in much greater detail in IAEA-TECDOC-978, constitutes a sound proof-of-principle for the PBR fuel. These data are in turn supported by a larger body of data on a variety of coated-particle fuel designs, which further broadens the understanding of coated-particle fuel behavior. International information exchanges and collaborations supported the successful transfer of important aspects of the German fuel process to other countries, and the experience of these countries provides a basis for confidence that the process can be successfully transferred to a PBR.

F.1.2.4 Planned additional irradiations

The German fuel program was terminated in the early 1990s, but archived fuel specimens are still available. The European Commission (EC) EURATOM RTD Framework Program first irradiation test is planned in the High Flux Reactor (HFR) on pebbles from the last German high-quality fuel production with the objective of reaching a burnup of 20,000 MWd/t. In the longer term, the EC program is planning fuel process development activities that will also include irradiations.

Large-quantity fuel irradiation is continuing in Japan with the operation of the HTTR. Follow-on HTTR core loadings are expected to be conducted with a fuel designed for higher burnup having a 500-μm kernel. Irradiation testing of this revised fuel design is expected to be conducted in the HTTR in the near term.

Irradiation of the HTR-10 first core loading fuel samples is continuing in IVV-2M, expecting to reach the target burnup of 11% FIMA in 2003. Large-quantity irradiation of the INET-produced fuel is in process via the continuing operation of HTR-10.

The PBMR project is planning to irradiate preproduction fuel in both the SAFARI reactor in South Africa and the IVV-2M reactor in the Russian Federation. In the longer term, production fuel irradiation in both reactors is also planned.

An IAEA CRP on Advances in HTGR Fuel Technology Development was initiated in 2002. This CRP will provide a mechanism for obtaining the results of the planned irradiations discussed in exchange for U.S. data from irradiation and safety testing.

F.1.2.5 Accident condition testing

The conditions experienced by the fuel during accidents are determined by analysis of the plant response to design basis events and other events of lower probability that are considered relevant to emergency planning, typically designated emergency planning basis events. Events can be categorized as heatup events, associated with a loss of coolant with no active residual heat removal; oxidation events,

associated with air or water ingress at high temperature; and reactivity transients, associated with control rod motion or changes in core geometry. Existing international data of direct relevance to PBR concepts for these categories are summarized in the following sections.

F.1.2.5.1 Heatup testing

The irradiation history of the fuel prior to a heatup event has been shown to be an important factor affecting how the fuel will perform. In an actual PBR, the core will contain a mixture of fuel ranging from fresh fuel to fuel approaching the allowed burnup limit. Fuel samples used in heatup testing have spanned a range of burnups, but most of the data are associated with burnups approaching or exceeding the discharge burnup. Fuel performance during a heatup event is a function of time at temperature, with peak temperatures (typically approximately 1600°C) reached and a slow decline beginning over a period of several days. Much of the testing has been conducted by heating irradiated fuel specimens to a specific temperature and holding it constant for periods ranging from 30 to 500 hours. Testing has also been conducted that simulates the expected time-dependent temperature behavior, as well as with slowly increasing temperatures reaching as high as 2500°C.

Figure F.5 shows the range of temperatures used in constant-temperature post-irradiation heatup tests in Germany and Japan. In addition to the heatup tests, a limited amount of irradiation testing at high temperatures (1500–2000°C) was conducted in Japan. The behavior of radionuclides during the tests is determined by on-line gaseous release measurements, periodic metallic fission product release measurements using a cold finger in the furnace, and post-test fission product profiling of the test specimens and holders. The Russian program included high-temperature irradiations, as reported earlier, and linear ramps to temperatures in excess of 3000°C. China will be conducting heatup tests of fuel samples currently under irradiation in the IVV-2M reactor in Russia.

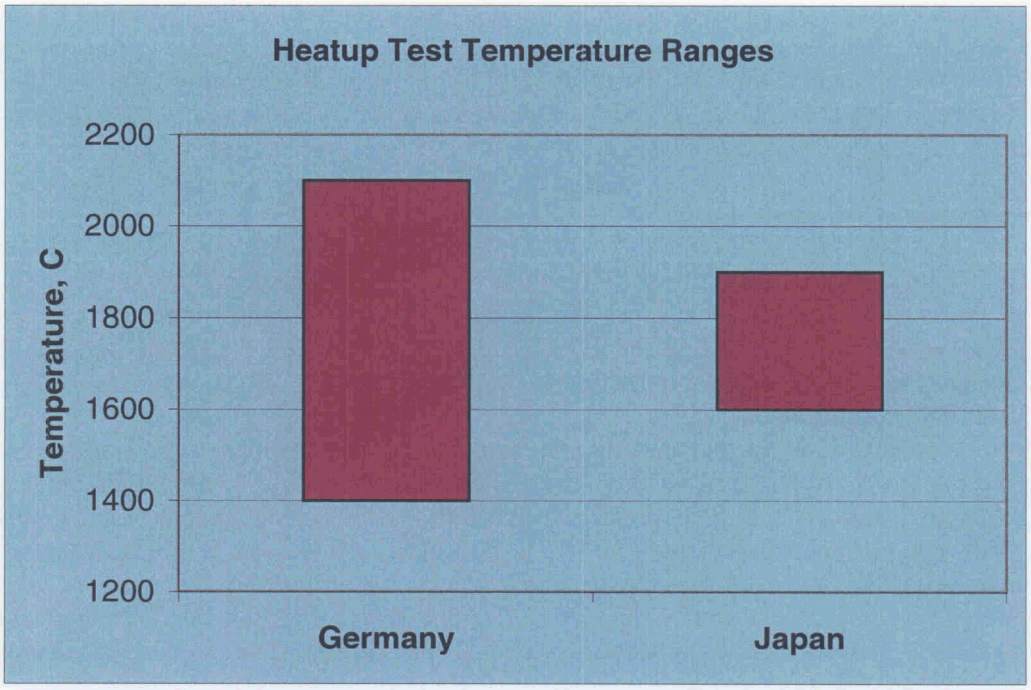

Fig. F.5. Heatup test temperature ranges.

F.1.2.5.2 Oxidation testing

Oxidation of graphite and coated-particle fuel has been addressed since the beginning of HTGR technology development over 40 years ago. For the steam cycle designs, oxidation from steam or water entering the primary system as a result of steam generator tube leaks has been the most likely issue. In the case of the gas turbine designs, there is no steam generator and the water heat exchangers operate with much smaller inventories at pressures far below the helium pressure; so significant water in-leakage during power operation is highly unlikely.

Air ingress following a system depressurization has also been studied in considerable detail. Significant oxidation due to air ingress is conceivable in the case of a very large break that remains for several days without mitigating action. However, such an event is sufficiently unlikely that it is typically not considered to be within the PBR licensing basis. Nonetheless, considerable data exist with regard to oxidation of UO_2 TRISO coated-particle fuel, as summarized below.

Extensive international testing has been conducted with regard to oxidation resulting from moisture ingress. These tests addressed conditions that could occur with large amounts of water entering the primary system under hot pressurized conditions, diffusing into the fuel elements, and attacking the particles. Earlier testing had shown that fission product retention by intact coated particles is not affected by the moisture, so the tests focused on failed particles. Some tests involved designed-to-fail particles (e.g., thin coatings with missing buffers), while others involved locating failed particles in irradiated fuel elements or crushing particles deconsolidated from the elements. The release characteristics for failed particles as a function of burnup and temperature are reasonably well understood.

Testing for oxidation resulting from air ingress has also been conducted. In Japan, unirradiated particles and compacts were subjected to an air atmosphere for temperatures ranging from 900 to 1400°C. The results showed a low level of particle failure (5.4×10^{-4} failure fraction) in loose particles exposed to air for 600 hours at 1300°C. Fuel compact behavior was quite different, with the difference attributed to the possibility of elevated temperatures occurring in the compact interior. In Germany, irradiated particles and spheres were subjected to an air environment for temperatures ranging from 1300 to 1620°C. Loose particle tests showed failures increasing to 100% at 1500°C. Sphere testing showed low failures ($\sim 10^{-4}$ failure fraction after 400 hours) at 1300°C, increasing ($\sim 10^{-3}$ failure fraction after 140 hours) at 1400°C. These data indicate a high degree of retention of fission products even under oxidizing conditions where all of the graphite outside the silicon carbide coatings has been consumed.

F.1.2.5.3 Reactivity transient testing

The online refueling of a PBR allows for operation with a limited amount of excess reactivity, thus limiting the potential for reactivity transients. Within the PBR design and licensing bases, reactivity transients are typically relatively benign and the resulting fuel temperature conditions are well within temperatures addressed for heatup events. A limited amount of reactivity transient testing of UO_2 coated-particle fuel has been conducted.

Short-term pulse tests were conducted in both Japan and Russia. In Japan, loose particles and compacts were subjected to pulses of 10–30 millisecond duration, with energy deposition ranging from 200 to 2300 J/gUO_2. In Russia, loose particles and spheres were subjected to pulses of 1–2 millisecond duration, with energy deposition ranging from 100–1700 J/gUO_2. The results of these tests are in good agreement and are widely available in summary form.

In addition to the pulse tests, longer-duration high-power tests were conducted on spherical fuel elements in Russia. In the first series of tests, the fuel elements were subjected to three sequential power pulses: 1.6 seconds at 150 kW/element, 1.0 seconds at 300 kW/element, and 0.7 seconds at 620 kW/element—values exceeding 100 times the design maximum power per sphere. In the second series of tests, the fuel elements were subjected to three pulses ranging in duration from 7 to 30 seconds at a power level of 46 kW/element, approximately 10 times design power levels. The most extreme of

these tests reached maximum temperatures of approximately 3000°C and resulted in the destruction of the spheres, while the spheres remained intact in less extreme tests. These data illustrate the capacity of the fuel to withstand large overpower events.

F.2 MARGIN TESTING CONSIDERATIONS FOR THE AGR FUEL DEVELOPMENT AND QUALIFICATION TESTING PROGRAM

The purpose of margin testing is to increase understanding of the limits of the fuel in terms of the key operating parameters (e.g., temperature, burnup, and fast neutron fluence) and the failure mechanisms [e.g., internal overpressure, inner pyrolytic carbon (IPyC) cracking, kernel migration, and fission product attack] for the fuel forms under consideration. Margin tests can give confidence that there is adequate margin in the fuel performance capability relative to uncertainties in the establishment of and compliance with normal operation and accident design limits. Margin testing at higher temperature, higher burnup, and higher fast neutron fluence relative to design limits can assist in determining the importance of potential accident sequences that may be postulated as part of the design basis and can provide the understanding necessary to justify the limited emergency planning zones proposed for AGRs. However, for the VHTR, design limits are not firm at present and are based on the designer's expectations of fuel performance capability using currently available data; thus, there is an iterative element to the concept of margin testing. A key issue for the program is the degree to which these variables are to be pushed in the irradiation and safety testing. Addressing this issue effectively requires a thorough understanding of existing coated-particle fuel performance data and their relevance to the fuel forms being developed and qualified by the program.

F.2.1 General Background

Adequate margins to failure of systems, structures, and components are necessary from the viewpoint of the designer (and ultimately the owner/operator) of a nuclear plant, as well as the perspective of regulatory agencies. The owner/operator has responsibility for the reliable and economical operation of the plant, as well as the primary responsibility for safety of the operating staff and the general public; regulatory agencies have oversight responsibilities to ensure that the responsibilities of the owner/operator are adequately addressed. Margins can be divided into two categories:

1. Margin between expected operating conditions and specified acceptable design limits—This margin addresses variations in normal operating state variables, uncertainties in calculational models (e.g., 95% probability and 95% confidence values), control and instrumentation errors, instrument drift, and control overshoot. It is established by plant designers with review and concurrence by the regulator, often in the form of industry codes and standards, with the intent of ensuring a very low probability of plant shutdowns or power reductions resulting from protection system actions and even an lower probability that specified acceptable design limits will be exceeded.

2. Margin between specified acceptable design limits and system, structure, or component failure—This margin addresses residual uncertainties in calculational models and in identification and characterization of events. It is established by plant designers with review and concurrence by the regulator, often in the process of developing industry codes and standards, with the intent of providing further assurance that the consequences of events that may occur at the plant will remain within acceptable limits.

Margin considerations for the testing program are based on category 2. Margins are discussed in several places in 10 CFR 50 (*Domestic Licensing of Production and Utilization Facilities*). The most relevant parts follow:

- 10 CFR §50.34(a)(3): *The preliminary design of the facility including: ... (iii) Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety.*
- 10 CFR §50.34(a)(4): "*A preliminary analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of (i) the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and (ii) the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents.*"

An example of explicit guidance regarding margins in 10 CFR 50 is the provisions for loss-of-coolant accidents for water reactors. Requirements for analysis methods described in 10 CFR 50, Appendix K, include detailed directions for conservative models and for assumptions regarding initial conditions. Acceptance criteria addressed in 10 CFR §50.46 include peak cladding temperature, maximum local cladding oxidation, and maximum total hydrogen generation. These criteria are directed toward precluding an autocatalytic Zircaloy-water reaction, maintaining coolable core geometry, and limiting hydrogen evolution to the containment; and they were based on a conservative assessment of the data and analyses available at the time. Note that the word "margin" does not appear in either Appendix K (addressing category 1) or §50.46 (addressing category 2). While it is possible to quantify the margin between best-estimate results (e.g., peak cladding temperature, and maximum cladding oxidation) for a specific plant and set of initial conditions, and the results of applying Appendix K methods, only a few margin requirements are quantified (e.g., 1.02 times the licensed power level). Margins between §50.46 limits and an autocatalytic Zircaloy-water reaction or loss of coolable geometry are not quantified.

F.2.2 Program Background

In 2001 and early 2002, Exelon Corporation conducted preliminary activities directed toward construction of PBRs (PBMR design) in the United States. The PBMR design was based on fuel that replicated the modern German fuel as developed and tested in Germany in support of the HTR-MODUL design developed in the 1980s. Preapplication interactions with the Nuclear Regulatory Commission (NRC) on a range of topics, including fuel, were a major element of these activities. Planning was in process to conduct irradiation and safety testing of PBMR fuel in the United States in support of PBMR deployment. In discussions with the NRC on fuel qualification irradiation and safety testing, a primary topic was demonstration of sufficient margins in fuel performance relative to operating and accident limits.

In April 2002, Exelon announced the termination of its initiative to deploy the PBMR in the United States and its withdrawal from the PBMR project in South Africa. In light of continued interest and support for development of modular HTGR technology as an option for future deployment in the United States, planning for coated-particle fuel irradiation and safety testing is being redirected to the broader goal of supporting future deployment of an AGR for commercial energy production in the United States by reducing market entry risks posed by technical uncertainties associated with fuel production and qualification. In the early stages, the redirected program is intended to support both pebble bed and prismatic designs, with the effort dedicated to each design proportionate with its associated level of industry interest and commitment. Because the pebble bed designs are based on replication of the well-established German fuel process, fuel fabrication process development will focus on prismatic fuel.

Most future prismatic fuel reactor concepts in the United States are based on the steam-cycle modular HTGR (MHTGR) concept funded by DOE in the 1980s and early 1990s. Little core design

activity specific to the GT-MHR was conducted before termination of the program in 1995, and the GT-MHR plant design has been pursued for initial application in Russia for consumption of surplus weapons plutonium, using a plutonium core. The GT-MHR, as a candidate for near-term deployment, would be expected to utilize LEU UCO fuel, as was the case for the MHTGR design. While conceptual designs are not available for a prismatic VHTR, it is expected that these concepts would also likely be initially fueled by LEU UCO fuel. Irradiations conducted on U.S.-produced LEU UCO fuel resulted in gaseous radionuclide releases well above the levels specified for the MHTGR and anticipated to be required for the GT-MHR, and approximately 3 orders of magnitude higher than levels observed for German LEU UO_2 fuel.[4]

Current pebble bed design activity in the United States is limited to specific technical issues being addressed at Massachusetts Institute of Technology. The PBMR design continues to evolve, with the objective of construction of a prototype in South Africa; pending design changes (e.g., a solid central reflector) are expected to substantially alter the fuel pebble service conditions. There is no integrated effort to advance a commercial GT-MHR design at present, so tentative service conditions are drawn from data developed under the earlier DOE-sponsored program. As a result, there is no stable and well-established set of fuel design limits for either concept at this time.

F.2.3 Discussion

Given the lack of adequate demonstration of satisfactory fuel performance (both normal and accident) in the United States over the past 25 years, the AGR program has decided that such a demonstration is critical to the feasibility of VHTR. Thus margin testing has a lower priority. The role of irradiation testing is to determine the capability of the fuel in terms that are useful to the reactor design and to licensing personnel. Irradiation testing should provide facilities with appropriate capability and use them to determine the responses of the fuel to the relevant in-service environmental parameters over appropriate ranges. Initial ranges of fuel testing conditions will be specified based primarily on expectations of fuel performance capability derived from existing data. These test ranges will be an integral part of the irradiation test specification to be produced prior to each irradiation. At this time, the providers of the irradiation testing capability can only anticipate what these requirements might be, design irradiation facilities to produce the expected conditions, and provide for the necessary measurements of environment and performance.

Based on analyses of existing coated-particle fuel irradiation and safety testing data and available fuel performance models, there is general consensus regarding the service conditions of primary importance with regard to fuel performance. In normal operation, they include temperature history and fast fluence and burnup, with lesser dependence on temperature gradients, fast flux, and power density. Temperature history is the primary service condition of interest for accident conditions; steam or air oxidation and transient overpower are of possible importance, depending on the provisions of a specific design.

Category 1 performance margins (as defined in the "General Background" section) will be established in the course of designing and licensing a plant. The margins of interest here are those of category 2. Establishment of category 2 performance margins for coated-particle fuel requires addressing the following areas:

- releases during normal operation as a function of service conditions exceeding design limits
- releases during accident conditions as a function of service conditions in normal operation exceeding design limits
- releases during accident conditions as a function of conditions experienced during the accident exceeding design limits

In the case of the LEU UO_2 pebble bed fuel, considerable data exist in all of the above areas to support estimates of failure thresholds by a designer, which can be used to establish design limits and corresponding margins to be confirmed by irradiation and testing of production fuel.[1,5] More limited data are available for LEU UCO prismatic block fuel, and many of the data are not supportive of the performance requirements anticipated for future prismatic concepts. Design limits for the MHTGR and future prismatic concepts established to date are based primarily on the presumption that LEU UCO fuel can be made to perform in a manner similar to the performance indicated by existing LEU UO_2 data.

It is important to keep in mind that HTGR fuel has no known autocatalytic reactions or propagation mechanisms equivalent to the Zircaloy-water reaction that could initiate in a hot region and propagate through the core if peak accident temperatures were higher than predicted, leading to a large-scale release of radionuclides. Thus, the radionuclide release during accident conditions that exceed predicted levels (e.g., 1600°C peak fuel temperature during a depressurized conduction cooldown) may be more dependent on a more substantial fraction of the fuel at a somewhat lower temperature than on the small fraction near the peak. Similar observations can be made regarding the distributions of operating temperature history, fluence, and burnup in the core. Quantification of margins to dose limits, which depends on the integrated release from the core, may be more dependent upon accurate knowledge of performance within the design limits than on performance at the extremes of the distributions, even if the extremes significantly exceed the design limits. Thus, the program should not overemphasize margin conditions at the expense of obtaining data within the design range.

Ultimately, the question of acceptable margins must be resolved between the plant designer and the regulator in the course of establishing specified acceptable design limits. The results will be predicated on the quantity and quality of data on fuel performance at and beyond the agreed limit. Optimally, the available data will cover a sufficient range to include the onset of significant failures to maximize the utilization of the capability of the fuel. Establishing the appropriate range of test conditions should be based on the best available understanding of how the fuel is likely to perform and where it is likely to begin to fail. Because existing plant concept design limits are based on expectations of fuel performance, they are an important input to the initial determination of the appropriate range of test conditions.

Note that more than 40 years and 400 units after the first water reactor demonstration plants, plant power uprates are still in process to take advantage of advancing knowledge of the performance capability of fuel and other components. It is extremely unlikely and unnecessary that the ultimate capability of the fuel in the first MHTGR demonstration plant will be established and exploited based on the results of this fuel development and qualification program. If a demonstration plant can be successfully deployed, large-scale data and experience with coated-particle fuel will be gained in support of the development and deployment of future plants.

F.2.4 Recommendations

The following recommendations are based on the background and discussion provided previously:

1. The near-term focus of the AGR fuel development and qualification program will be on the development of LEU UCO prismatic block fuel. Obtaining feedback on the irradiation and safety testing performance of early fuel products is highly desirable, and it may lead to the use of one or more "survey capsules" containing variants of key process or product parameters as candidates for the reference fuel. In this capsule or capsules, the conditions should remain within the design limits and not attempt to address margins.

2. Irradiation of reference fuel should be primarily directed toward providing statistically sufficient data on fuel performance within the design limits to support modeling of the integrated core behavior. In principle, it should be possible to perform the margin testing all in one capsule. Because the margin testing is aimed at exploring beyond the operating envelope, looking for failure mechanisms, the use of a large number of fuel specimens is expected to be unnecessary.

3. Test conditions for individual irradiation capsules will be established in the course of developing the test specification for the capsule. Testing for margins will be an important consideration in establishing the conditions at that time. Current considerations of margins should be limited to establishing an initial position, with the objective of providing reasonable assurance that initial test capsule designs will be capable of meeting the requirements for margin testing.
4. Table F.4 provides preliminary data on maximum values for the parameters of primary interest for fuel irradiation, based on input from the GT-MHR and PBMR designers. These data can be used for planning purposes, as the starting point for considerations of establishing irradiation and testing capabilities, and for further discussion regarding inclusion of margins in the testing. Final data for a given test will be provided in the test specification.

Table F.4. Required irradiation capabilities

Limiting test capability requirement	GT-MHR	PBMR[a]
Test samples		
Samples of individual loose particles in capsule intended for irradiating fuel bodies (piggy-back samples)	X	X
Special fuel material samples in capsules intended for irradiation of fuel bodies (piggy-back samples)	X	X
Fuel bodies	Compacts 12.5 mm × 50 mm	Spheres 60-mm diam
Statistically significant numbers of particles	X	X
Multicell capsules	X	X
Test environment		
Maximum fuel temperature	1350°C	1300°C
Maximum burnup	27% FIMA	11% FIMA
Maximum average burnup acceleration factor	3 (Maximum burnup in 12 months)	
Maximum fast fluence	6×10^{25} neutrons/m^2 (E > 0.18 MeV)	2.5×10^{25} neutrons/m^2 (E > 0.1 MeV)
Maximum average fast neutron fluence acceleration factor	3 (Maximum fluence in 12 months)	
Maximum fuel enrichment (%^{235}U)	20%	9%
Instantaneous maximum power/particle	0.2 W/particle	0.2 W/particle

[a]PBMR data derived from E. J. Mulder, "PBMR: Opting for an annular reactor layout," presented at IAEA 17th International Working Group on Gas Cooled Reactors, November 2001.

F.3 ACCELERATED IRRADIATION TESTING OF TRISO COATED-PARTICLE FUELS

Because of the differences in neutron flux spectrum between a gas reactor and a light-water materials test reactor, simultaneous matching of both the rate of burnup and the rate of accumulation of fast neutron fluence is difficult to achieve. In addition, the traditional 3-year fuel cycle of HTGRs makes real-time irradiation testing time-consuming and expensive as part of an overall fuel development effort. To overcome these shortcomings, irradiations in material test reactors have historically been accelerated relative to those in the actual reactor. Usually, the time acceleration is focused on achieving the required

burnup in a shorter time than in the actual reactor, with the value of the fast fluence left as a secondary variable that must fall between a minimum and maximum value.

The level of acceleration also can impact the potential for fuel failure during irradiation. The level of acceleration at a given test reactor power, coupled with fuel loading in the experiment, results in a power density for the fuel specimen in the experiment. The power density peaks at the beginning of the irradiation when the fissile content is highest and decreases as the fissile material is burned out of the fuel. The power density in the fuel specimen defines the temperature gradient across the fuel specimen and thus across the individual coated particles in the fuel body. Both fission product attack and kernel migration phenomena in coated-particle fuels have been found to be functions of the thermal gradient in the fuel body. As the thermal gradient increases, the potential for these failure mechanisms to play a deleterious role in fuel performance becomes more important.

To date, the linkage between the level of acceptable acceleration and satisfactory fuel performance is empirical at best.[1] In the German irradiation program, limits were placed on the peak power per pebble (~2500 W) to limit the thermal gradients. This limit corresponded to a level of acceleration of 1.5 to 3. By contrast, in the United States, the irradiations were accelerated between factors of 3 and 10, and the irradiation performance of United States fuel was much worse than in the corresponding German experience. The level of acceleration was one of many factors that were identified as important contributors to the poor U.S. irradiation performance.[3]

As a result, it would appear that modest acceleration (1.5 to 3X) appears to be acceptable without jeopardizing fuel performance in the irradiation, and it should be a baseline requirement in any future gas reactor irradiations. This acceleration level should be translated into a maximum power per fuel body or power per particle that can be used by experimenters in the design of the irradiation capsule. Meeting this requirement along with the other requirements of the irradiation can at times be difficult. Three options are available to meet this constraint:

1. irradiation in a location that closely mimics that expected in the gas reactor (e.g., somewhat higher flux levels but the same or similar spectrum, resulting in a slight acceleration of both burnup and fluence);
2. irradiation initially in a low-flux region of the materials test reactor and, following significant depletion of the fissile content, irradiation in a higher flux position; or
3. irradiation in a high-flux location of a materials test reactor with a thermal shroud that would reduce the flux early in the irradiation and that could then be removed later in the irradiation when the fissile density is low enough to meet the power constraint.

All three options have been or are under consideration in this program. Option (1) appears feasible for fuel compacts using a large B hole in the ATR. Option (2) was used for the New Production Reactor (NPR)-1 and NPR-2 experiments in the High-Flux Isotope Reactor at Oak Ridge National Laboratory. A combination of options (2) and (3) may be needed for irradiation of fuel pebbles in the ATR. Each of these options will result in a unique trajectory in the rate of accumulation of burnup and fast fluence. Thus, in any of these options, it is important to compare a plot of burnup vs fast fluence for the experiment with that expected in the reactor so that all involved understand the trade-offs associated with each option and make the most informed decision about the irradiation under consideration.

F.4 IMPORTANT RISK FACTORS ASSOCIATED WITH DEPLOYMENT OF A GCR SYSTEM

An important goal of the AGR program is to reduce the deployment risk to potential industrial participants to the point where they feel comfortable investing in the technology. The AGR program addresses fuel manufacturing and testing capability. Fuel manufacture is a large risk area; a number of

other risks have been addressed in previous DOE-sponsored HTGR development programs but are not addressed by the AGR program. The other important risk areas are discussed briefly below.

Support for a Demonstration-Scale Reactor: Nuclear utilities think that the data obtained from demonstration-scale plants is essential to support a decision associated with full-scale deployment of new technology. The demonstration-scale plant will provide needed information on costs of construction, fuel cycle costs, performance of components, and licensing issues. In addition, much of the data needed to support licensing of full-scale plants can best be obtained from demonstration-scale reactors. For example, fission product behavior in the primary circuit during normal and off-normal conditions is often difficult to obtain from test reactors and fission product loops. Further, integrated operations are required to validate fuel performance and fission product transport models.

Development of the Power Conversion System: A direct-cycle plant will require helium-driven turbomachinery. Helium turbines are not routinely used in commerce, and there are few data on performance, maintenance, and reliability. If helium-driven power conversion machinery proves to be inadequate for the application, it will be necessary to use air turbines and incorporate a helium-to-air heat exchanger in the design.

Reliability of Key Components: The Fort St. Vrain reactor experienced low online availability because of severe problems with developmental components, such as the helium circulator. New designs using unproven components often experience significant outages.

Availability of Code-Qualified Materials of Construction for Primary Circuit: Modified 9Cr–1 Mo steel is high-temperature *ASME Code* nuclear-qualified, but this material is not sufficiently refractory to withstand the 850°C outlet temperatures of the GT-MHR or PBMR. Allowable stresses for 9Cr–1 Mo are included in the Code case up to 1200°F (640°C), and fatigue curves are available only to 1000°F (538°C). In the late 1980s, a draft nuclear Code case was developed for Alloy 617 for temperatures up to 900°C, at the request of General Electric (GE). The Code case was moving through the Code approval chain, but GE lost its HTGR funding, and the effort was terminated. There is interest by both prismatic and pebble bed reactor designers in carbon-carbon composite materials, but nothing has yet been submitted for Code qualification. The quickest and least expensive pathway to a Code-qualified material would most likely be a resumption of the review effort for Alloy 617.

Availability of Nuclear-Grade Graphite:

- Previous programmatic efforts (in the 1970s) for the HTGR have "qualified" H-451 graphite for fuel element use. Here the term "qualified" refers to the experience and operational data required for use in competent nuclear design. H-451 was manufactured in the past by SGL, but SGL no longer manufactures this product. Given the cost and time required to "qualify" graphite, the most efficient path forward would be to establish another supplier of a product that behaves like H-451.
- Several companies [notably SGL, Graftek (formerly UCAR), and Toyo Tanso] currently have "nuclear graphite" available. All three bid on the PBMR project. To supply a clone of the 1970s-vintage H-451, it will be necessary to choose a new coke, make some graphite, and perform some irradiations to ensure that the new graphite behaves like the old graphite. An alternative to this course of action would be to identify the best currently available graphite and then generate the design database from scratch.

Fabricability and Transportability of the Reactor Vessel: Gas-cooled reactors gain significantly in the safety area through the lower specific power of the core (compared with light water reactors). The large core has some drawbacks, however. One is the requirement for a large reactor vessel (22 ft in diameter and >700 tons). There is little industrial experience with the manufacturing and transport of such large nuclear-grade components.

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