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Deep-Burn Modular Helium Reactor Fuel Development Plan

September 2002

Prepared by

General Atomics
Oak Ridge National Laboratory

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General Atomics
and
Oak Ridge National Laboratory

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Acronyms and Abbreviations

Acronym	Definition
AC	accident condition
AD-MHA	Accelerator Driven-Modular Helium Assembly
AGR	Advanced Gas Reactor
Am	americium
ATR	Advanced Test Reactor (INEEL)
AVR	Arbeitsgemeinschaft Versuchs Reaktor, a German experimental reactor
BAF_o	Bacon Anisotropy Factor
BISO	Coated-fuel particle design with two materials in coating system (low-density PyC and high-density PyC)
CCCTF	Core Conduction Cooldown Test Facility
CH-TRU	contact handled transuranic
Cm	curium
Cs	cesium
CVD	chemical vapor deposition
DDN	Design Data Need
DB-MHR	A GT-MHR used to destroy TRU from LWRs
DF	Driver Fuel
DTF	“designed-to-fail”
DOE	[United States] Department of Energy
FDDM	Fuel Design Data Manuel
FDP	Fuel Development Plan [i.e., this document]
FG	fission [product] gases
FIMA	fissions per initial metal atom [a measure of burnup in fuel]
FM	fission [product] metals
FSV	Fort St. Vrain
FZJ	Forschungszentrum Juelich
GA	General Atomics
GmbH	[German equivalent of the English abbreviation “Inc.”]
GT-MHR	Gas-Turbine Modular Helium Reactor (Brayton-cycle plant)
HEU	Highly-enriched uranium (usually ~93%)

Acronym	**Definition**
HFIR	High Flux Isotope Reactor (ORNL)
HTGR	High Temperature Gas-Cooled Reactor [generic term for reactor type]
HTR	High Temperature Reactor [pebble-bed core]
HTTR	High Temperature Test Reactor
INEEL	Idaho National Engineering and Environmental Laboratory
IMGA	Irradiated Microsphere Gamma Analysis, a test at ORNL
IPyC	inner pyrolytic carbon coating in a TRISO particle
kw	kilowatt
LF	laser failed
LWR	light-water reactor
MB	missing buffer
MHA	sub-critical assembly of MHR fuel blocks driven by an accelerator
MHR	Modular Helium Reactor
MW	megawatt
NOC	normal operating condition
NPR	New Production Reactor, an application for producing tritium
OPyC	outer pyrolytic carbon coating in a TRISO particle
ORNL	Oak Ridge National Laboratory
PBMR	Pebble Bed Modular Reactor
QC	quality control
SEM	scanning electron microscope
REDC	Radiochemical Engineering Development Facility (ORNL)
RN	radionuclide
RTW	Reactor Transmutation of Waste program
SiC	silicon carbide coating in a TRISO particle
TBD	to be determined
TF	Transmutation Fuel
THTR	Thorium High Temperature Reactor
TRIGA	Test, Research, Isotopes General Atomics [a test reactor]
TRISO	TRi-ISOtropic coated-fuel particle design with three materials in coating system (low-density PyC, high-density PyC, and SiC)
TRU	transuranic nuclides

Acronym	**Definition**
TURF	Thorium/Uranium Recycle Facility at ORNL
UCO	a mixture of UO_2 and UC_2
UREX	uranium extraction processing method
WAR	Weak Acid Resin [kernels]
WBS	work breakdown structure

1. Executive Summary

High Temperature Gas-cooled Reactors (HTGRs) offer unique advantages for the destruction of transuranic materials in spent fuel discharged from commercial light water reactors. The Gas Turbine-Modular Helium Reactor (GT-MHR) is being evaluated for operation as an efficient burner of TRU materials, with some modification required for the reactor core design. The GT-MHR, in this application as a transmutation burner, is referred to as the Deep-Burn MHR (DB-MHR). The primary benefit of the DB-MHR concept is to reduce the long-term storage requirements for high-level waste generated by the currently operating nuclear reactors around the world and to facilitate the acceptance of new nuclear power plants to meet the world power demands.

The DB-MHR can be used to convert the transuranic radionuclides, recovered from spent LWR fuel, into shorter-lived fission products. The transmutation is accomplished first in a DB-MHR, using a plutonium/neptunium Driver Fuel (DF). In a single pass of DF through the DB-MHR, nearly all-fissile plutonium and much of the neptunium are destroyed by fission. The minor actinides from the reprocessed LWR spent fuel and the residual heavy nuclides recovered from the first-pass Driver Fuel are combined and made into a TRISO-coated Transmutation Fuel (TF).

The Transmutation Fuel is passed through the DB-MHR where 35 to 40% of the remaining transuranics are destroyed. When the TF is removed from the DB-MHR the TRU has been reduced to about 25% of its original mass. After irradiation the TF compacts are removed from the fuel element. At this point the spent Transmutation Fuel can be sent to the repository or can be loaded into new fuel assemblies and irradiated in an accelerated-driven subcritical assembly which reduces the mass of TRU to 5% of its original mass. (Unprocessed spent HTGR fuel elements with their ceramic-fuel particles and monolithic graphite blocks represent a nearly ideal waste form for permanent geologic disposal.)

This Fuel Development Plan describes part of the overall development program for the reactor plant and the supporting fuel facilities needed to transmute TRU stored in spent nuclear fuel. The Plan includes the work on fuel necessary to support the design and licensing of the DB-MHR, as well as the development of supporting facilities for fuel fabrication and spent fuel disposal. The scope of work addressed in the Plan can be grouped in four related areas: (1) fuel process development, (2) fuel materials performance, (3) fission product transport, and (4) spent fuel disposal.

Four major features of the thermal spectrum DB-MHR with its ceramic coated fuel particles and graphite moderator with are uniquely beneficial for TRU destruction: (1) possibility of high burnup of fissionable material in the fuel, (2) no requirement for added fertile material to provide negative temperature coefficient, (3) no requirement for burnable poison, and (4) the coatings on the fuel particles may provide confinement of the residual TRU material in spent fuel for ten of thousands of years or more. Because of the high burnup fuel and the excellent spent fuel integrity, the spent fuel recycling incentives and radionuclide losses are minimized.

The formal goal of the fuel development program is to satisfy the Design Data Needs (sS data, and providing the basis for the design of an industrial-scale fuel fabrication plant for supplying fuel to an NRC-licensed DB-MHR. At the conclusion of the program: (a) a preliminary design and cost estimate will have been prepared for constructing the fuel fabrication plant, and (b) the testing of prototype fuel will be complete. Sufficient data will be available to support a decision on detailed design and construction of a full-scale fuel fabrication plant and DB-MHR, and to support large-scale fuel-element irradiation testing in a prototype (or demonstration) reactor.

The development activities described in this plan are to be carried out primarily at General Atomics (GA) and Oak Ridge National Laboratory (ORNL). The program requires some unique facilities and equipment but also makes substantial use of facilities which were constructed within earlier National programs for nuclear power plant development.

The DB-MHR fuel development program will systematically coordinate its activities with other U.S. and international, coated-particle fuel development activities. Two on-going programs are of particular importance. First, the joint DOE-NNSA/MINATOM International GT-MHR program for the disposition of surplus Russian weapons plutonium is developing high-burnup, TRISO-coated Pu fuel which is quite similar to the Driver Fuel to be developed by this program. Secondly, the DOE-NE sponsored, Advanced Gas Reactor (AGR) fuel development program has been planned to develop and qualify LEU coated-particle fuel to be used in commercial PBMR and GT-MHR designs. Fuel development activities sponsored by the European Union, China, and Japan should also produce directly relevant data (e.g., the latter's planned development of advanced ZrC coatings).

The summary schedule for DB-MHR fuel development is presented in Fig. 1-1. This summary representation also shows the temporal relationship between the fuel development program and the establishment of a Demonstration Modular Helium Reactor. The schedule is based on beginning work in 2002 and achieving a capability for fabrication and testing DF and TF in five years at a cost of $85 million. Pre-production quantities of fuel of each type will be delivered within six years at a cumulative cost of $111 million. The total cost of the entire fuel development plan workscope is $216 million.

When the work included in this fuel development plan is completed, the following facilities and technical information will have been developed:

- Both Driver and Transmutation Fuels will be qualified for use in fuel test elements for transmutation in a DB-MHR; this includes:
  - Statistically significant irradiation data under normal operating conditions for Driver and Transmutation Fuels
  - Statistically significant irradiation data under accident conditions for Driver and Transmutation Fuels
  - Understanding of the thermochemical and structural behavior of Driver and Transmutation Fuels
- Information needed for the design and construction of a fuel plant to mass produce Driver and Transmutation Fuels will be available; this includes:
  - Process flow sheets for Driver and Transmutation Fuel fabrication
  - Procedures and operating parameters for fabrication

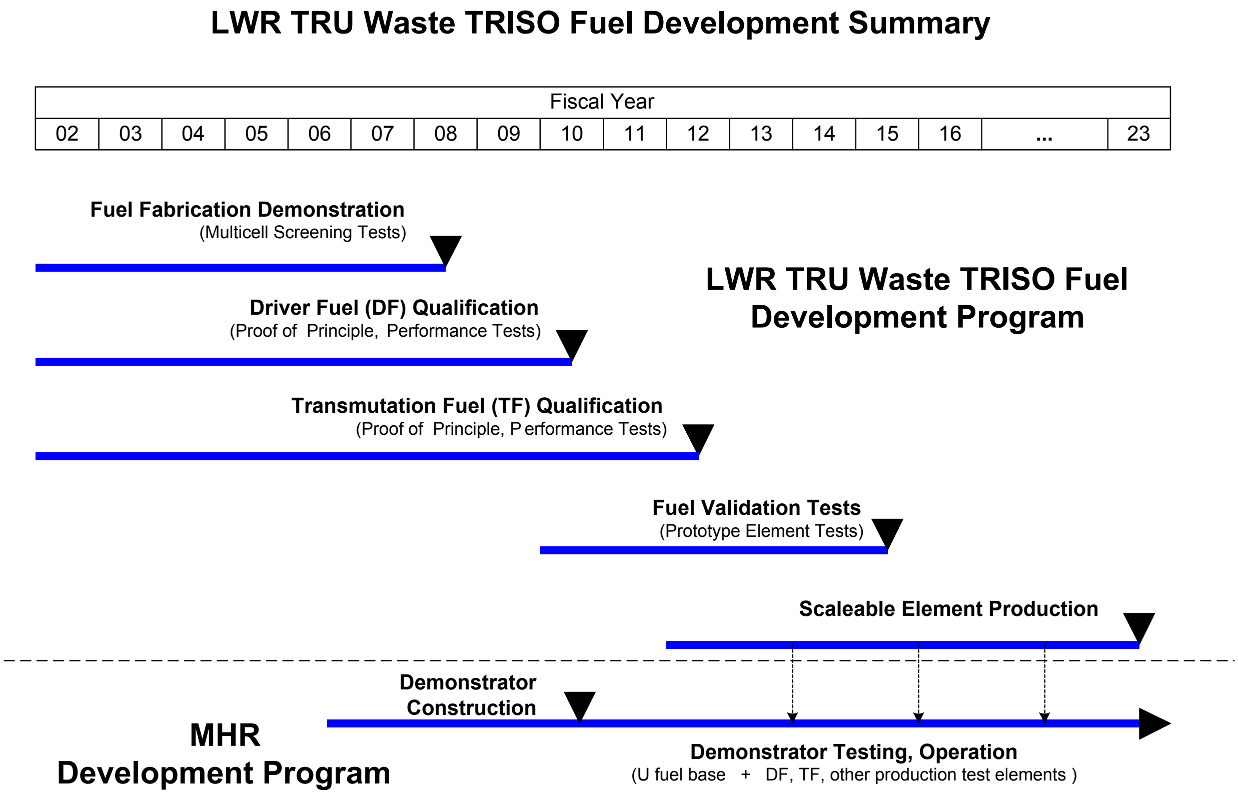


Figure 1-1. Summary Schedule for DB-MHR Fuel Development

  - conceptual design of the fabrication facility including a cost basis for the fabrication plant and the cost of the fabricated fuel

- Technical information needed to validate the fission product source terms for the DB-MHR will be produced for normal operation and accident conditions; this includes:
  - Fission product transport data for the kernels, coatings, compacts and fuel-element graphite as a function of irradiation conditions
  - Data for validation of fission product codes used for reactor design and safety analysis

- Although existing facilities are extensively used to accomplish the work of this plan, some new facilities will be established; these include:
  - Laboratory-scale facilities:
    - Facility for fabricating inert and uranium -containing TRISO particles and compacts and for process development
    - Glovebox facility to fabricate Driver Fuel kernels, coatings and compacts and to fabricate compacts containing small quantities of Transmutation Fuel
    - Hot cell facility to fabricate Transmutation Fuel kernels, coatings and compacts
    - Capability to characterize Driver and Transmutation Fuels
  - Capability to perform multi-cell capsule irradiations
  - Capability to perform corrosion and leaching tests on fresh and irradiated fuels

2. Introduction

2.1 Purpose

This document contains the workscope, schedule and cost for the technology development tasks needed to satisfy the fuel and fission product transport Design Data Needs (DDNs) for the Gas Turbine-Modular Helium Reactor (GT-MHR), operating in its role of transmuting transuranic (TRU) nuclides in spent fuel discharged from commercial light-water reactors (LWRs). In its application for transmutation, the GT-MHR is referred to as the Deep-Burn MHR (DB-MHR).

2.2 Background

2.2.1 Scope and Content of Plan

This Fuel Development Plan (FDP) describes part of the overall program being undertaken by the U.S. Department of Energy (DOE), utilities, and industry to evaluate the use of the GT-MHR to transmute transuranic nuclides from spent nuclear fuel. The Fuel Development Plan (FDP) includes the work on fuel necessary to support the design and licensing of the DB-MHR.

The FDP is organized into ten sections. Section 1 provides a summary of the most important features of the plan, including cost and schedule information.

Section 2 describes the DB-MHR concept, the features of its fuel and the plan to develop coated particle fuel for transmutation.

Section 3 describes the knowledge base for fabrication of coated particles, the experience with irradiation performance of coated particle fuels, the database for fission product transport in HTGR cores, and describes test data and calculations for the performance of coated particle fuel while in a repository.

Section 4 presents the fuel performance requirements in terms of as-manufactured quality and performance of the fuel coatings under irradiation and accident conditions. These requirements are provisional because the design of the DB-MHR is in an early stage. However, the requirements are presented in this preliminary form to guide the initial work on the fuel development. Section 4 also presents limits on the irradiation conditions to which the coated particle fuel can be subjected for the core design. These limits are based on past irradiation experience.

Section 5 describes the Design Data Needs to: (1) fabricate the coated particle fuel, (2) predict its performance in the reactor core, (3) predict the radionuclide release rates from the reactor core, and (4) predict the performance of spent fuel in a geological repository.

The heart of this fuel development plan is Section 6, which describes the development activities proposed to satisfy the DDNs presented in Section 5. The development scope is divided into Fuel Process Development, Fuel Materials Development, Fission Product Transport, and Spent Fuel Disposal.

Section 7 describes the facilities to be used. Generally, this program will utilize existing facilities. While some facilities will need to be modified, there is no requirement for major new facilities.

Section 8 states the Quality Assurance requirements that will be applied to the development activities.

Section 9 presents detailed costs organized by WBS and spread over time.

Section 10 presents a list of the types of deliverables that will be prepared in each of the WBS elements.

Four Appendices contain supplementary information on: (a) design data needs, (b) the interface with the separations plant, (c) the detailed development schedule, and (d) the detailed cost estimate.

2.2.2 The Deep Burn Concept

To meet world power demands, the use and acceptance of nuclear power must continue to grow as a safe, reliable, and economical source of energy. To that end, the current political impasse over the ultimate disposition of spent LWR fuel must be resolved. The rate of waste generation by the currently operating U.S. nuclear reactors would require that new repository capacity equal to the statutory capacity of the yet-to-open Yucca Mountain Geologic Waste Repository be added about every 20-30 years. Therefore, the ability to expand, or even maintain the nuclear power capacity in the U.S. may soon be limited, unless either additional disposal capacity is made available, or the volume, proliferation risk, and toxicity of the waste are significantly reduced.

Based on this motivation, methods to reduce nuclear waste volume and toxicity have been proposed. Critics have argued that methods to concentrate the waste or to expand additional disposal capacity would generate significant amounts of new waste, create new opportunities for plutonium diversion, require long deployment times, and prove to be very costly. One concept that offers great potential for improved reactor safety, attractive economics, and environmental compatibility is the DB-MHR with its coated particle fuel (Venneri 2001).

2.2.2.1 Deep-Burn Advantages

Deep-Burn transmutation is based on the use of thermalized neutrons and high burnup fuel forms in MHRs. These reactors have annular graphite-moderated cores and are designed to be passively safe at power levels up to at least 600 MW thermal. They can operate safely at very high coolant temperatures, resulting in electric power production at 50% plant efficiency or greater.

Helium-cooled, graphite-moderated reactors with ceramic-coated fuel particles offer unique advantages for the destruction of transuranic (TRU) materials discharged from Light Water Reactor (LWR) spent fuel. TRU destruction is accomplished by high burnup of fissile nuclides and also by capture, followed by fission, for non-fissionable isotopes. In the DB-MHR transmutation scheme proposed here, coated particle fuels offer the possibility of virtually complete destruction of weapons-usable plutonium, and 95% destruction of all transuranic waste

(Venneri 2001). Levels of destruction higher than 95% are possible by repeated reprocessing, refabrication of coated particles, and recycling through the system. However, there is little incentive for numerous recycles because each reprocessing step is costly and generates new secondary waste. Moreover, unprocessed spent HTGR fuel elements are nearly ideal final waste forms for permanent geologic disposal because their TRISO-coated particles and monolithic graphite blocks should contain the residual radionuclides for hundreds of thousands of years.

Four major features of the thermal spectrum DB-MHR are uniquely beneficial for TRU destruction:

1. High burnup of fissionable material in the fuel in a single pass
   a. Reduces the requirement for multiple reprocessing and refabrication steps
   b. Reduces the total amount of fissionable materials in the fuel cycle
   c. Produces spent fuel for the repository which contains essentially no fissionable isotopes
   d. Sharply reduces the requirements for separations and refabrication
   e. Reduces the requirements for tier 2 facilities
2. No requirement for added fertile material to provide negative temperature coefficient
   a. Negative temperature coefficient is provided by the non-fissionable TRU nuclides taken from the LWR spent fuel
   b. Therefore, neutrons are not used in fertile material which produces more Pu but are used to convert non-fissile TRU to fissile TRU thus furthering the transmutation process
3. No requirement for burnable poison
   a. The role of burnable poison is taken by the non-fissionable TRU nuclides
4. The coatings on the fuel particles should provide confinement of the residual TRU material for hundreds thousands of years or more
   a. Toxic radionuclides (RNs) are confined in the particle coatings in the repository, thus relaxing the requirement for high levels of destruction of TRU.
   b. Reduces the requirement for numerous recovery, refabrication, and recycle of fuel through the system
   c. Reduces the inventory of TRU material in the fuel cycle

2.2.2.2 Deep Burn Fuel Cycle

The DB-MHR can be used to convert the transuranic radionuclides, recovered from spent LWR fuel, into shorter-lived fission products either in a first tier reactor or in a two-tier system (Venneri 2001). Figure 2-1 shows the material flow through the two-tier system.

The transmutation is accomplished first in a DB-MHR, using a plutonium/neptunium Driver Fuel (DF). In a single pass of DF through the DB-MHR, nearly all-fissile plutonium and much of the neptunium are destroyed by fission.

The minor actinides from the reprocessed LWR spent fuel and the residual heavy nuclides recovered from the first-pass Driver Fuel are combined and made into a TRISO-coated Transmutation Fuel (TF). Because essentially no fissionable material remains in the spent DF (and there is little in the minor actinide stream), this separation and refabrication is free of

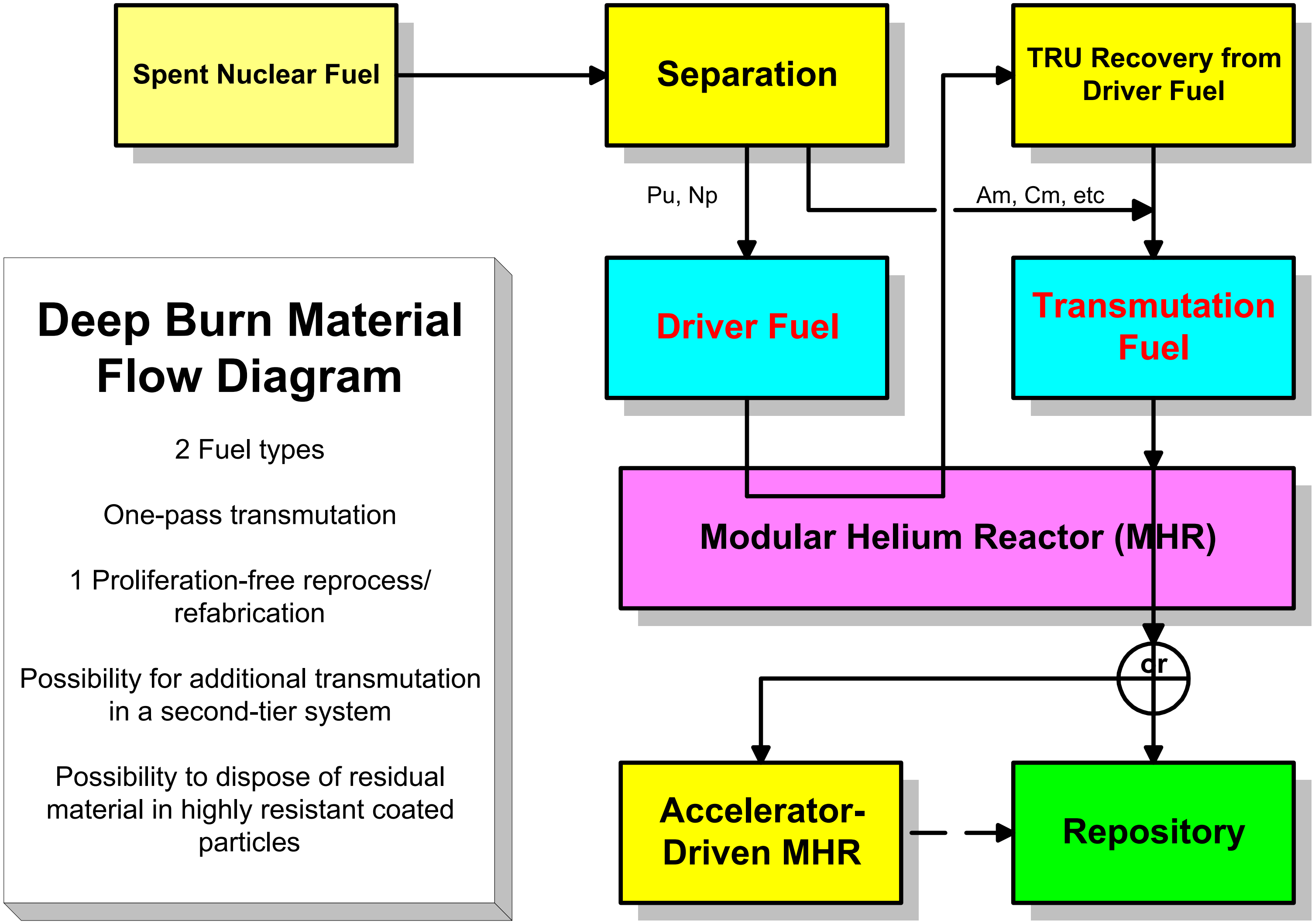


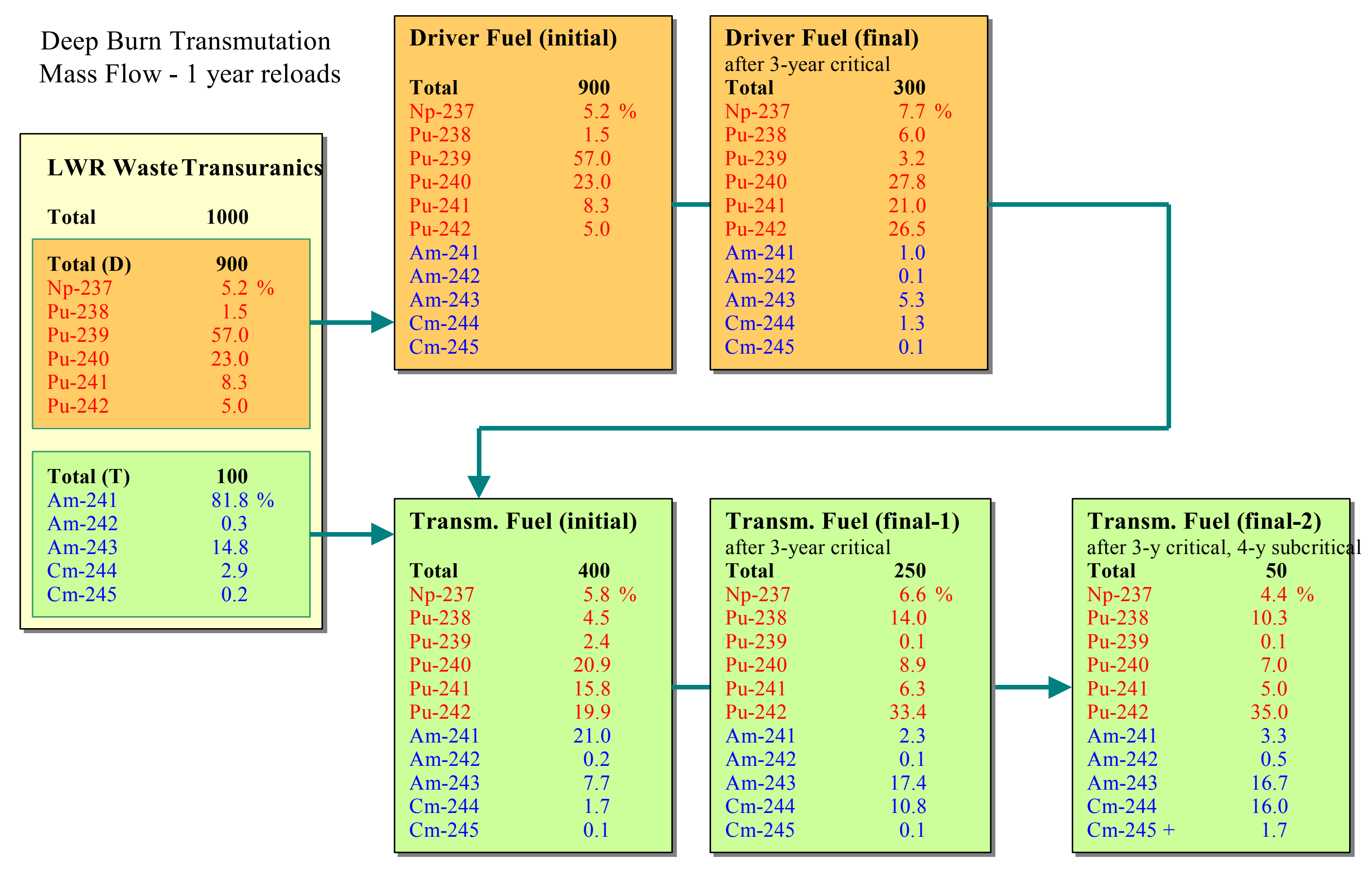
Figure 2-1. Flow of DF and TF through the DB-MHR

proliferation risks. Development work is needed to demonstrate the processes to make the separation of the transuranics from the spent Driver Fuel for use in the Transmutation Fuel. The necessary separations developmental work will be described in another development plan devoted to that subject.

The isotopic compositions of the TRU nuclides are shown at each stage of the fuel processing in Fig. 2-2. The mass and composition of 1000 kg of TRU from LWR fuel is traced as it is irradiated in the DB-MHR as DF, then as TF, and finally passes through the second tier Accelerator Driven-Modular Helium Assembly (AD-MHA). The mass flows are for a DF and TF residence time of three years in the DB-MHR and a three years residence time in an AD-MHA. The TF is passed through the DB-MHR where 35 to 40% of the remaining transuranics are destroyed. When the TF is removed from the DB-MHR the TRU has been reduced to about 25% of its original mass. After irradiation the TF compacts are removed from the fuel element. At this point the spent Transmutation Fuel can be sent to the repository or can be loaded into new fuel assemblies and irradiated in the AD-MHA (tier 2 system) which reduces the mass of TRU to 5% of its original mass (McEachern 1999). In principle, the TF could be reprocessed multiple times and the remaining transuranics recovered and made into TF and irradiated again in the DB-MHR and the AD-MHA to obtain further destruction of the transuranics. However, only one pass through the AD-MHA was chosen as a reference, because it is important to keep the number of cycles of separation of transuranics and fuel refabrication cycles low to avoid the cost and additional waste generated in multiple reprocessing and to minimize the in-process inventories of transuranics.

Either when removed from the DB-MHR as spent TF (one tier transmutation) or after passing through the AD-MHA (two tier transmutation), the spent TF is sent to the repository. The TRISO coatings provide an exceptionally long-lived barrier to confine the remaining material in the repository. The TRISO coating system on the TF is an excellent engineered barrier for containing radionuclides when the TF is transferred to a geologic repository. In the early 1980s, ORNL developed coated particle waste forms based on TRISO coatings (Stinton 1982). Microspheres containing simulated nuclear waste were coated either with only PyC or with several combinations of PyC and SiC. Comparative leach tests were performed on the coated waste particles and with more conventional glassified waste forms. For the coated waste particles, radionuclides were not detected in the leachate using sensitive analytical techniques. In contrast, radionuclide release was readily detected from glassified waste forms.

Also in the early 1980s, the corrosion behavior of graphite, glassy carbon pyrocarbons, and SiC was investigated by the Pacific Northwest Laboratory (PNL) to evaluate improved barriers for nuclear waste isolation (Gray 1982). Their data showed that the graphite and TRISO coatings should maintain their integrity for hundreds of thousands of years in a dry or wet repository environment. The ORNL and PNL data showing excellent retentive properties were consistent with data obtained in Germany (Kirch 1990). The German researchers tested the leach resistance of TRISO-coated particles at temperatures to 150°C and pressure up to 200 atmospheres in brine for exposure times exceeding four years. The TRISO coatings showed no indication of corrosion. Calculations have been performed to predict integrity of discharged TRISO coated



Deep Burn Transmutation
Mass Flow - 1 year reloads

LWR Waste Transuranics

Total	**1000**
Total (D)	**900**
Np-237	5.2 %
Pu-238	1.5
Pu-239	57.0
Pu-240	23.0
Pu-241	8.3
Pu-242	5.0
Total (T)	**100**
Am-241	81.8 %
Am-242	0.3
Am-243	14.8
Cm-244	2.9
Cm-245	0.2

Driver Fuel (initial)

Total	**900**
Np-237	5.2 %
Pu-238	1.5
Pu-239	57.0
Pu-240	23.0
Pu-241	8.3
Pu-242	5.0
Am-241	
Am-242	
Am-243	
Cm-244	
Cm-245	

Driver Fuel (final)
after 3-year critical

Total	**300**
Np-237	7.7 %
Pu-238	6.0
Pu-239	3.2
Pu-240	27.8
Pu-241	21.0
Pu-242	26.5
Am-241	1.0
Am-242	0.1
Am-243	5.3
Cm-244	1.3
Cm-245	0.1

Transm. Fuel (initial)

Total	**400**
Np-237	5.8 %
Pu-238	4.5
Pu-239	2.4
Pu-240	20.9
Pu-241	15.8
Pu-242	19.9
Am-241	21.0
Am-242	0.2
Am-243	7.7
Cm-244	1.7
Cm-245	0.1

Transm. Fuel (final-1)
after 3-year critical

Total	**250**
Np-237	6.6 %
Pu-238	14.0
Pu-239	0.1
Pu-240	8.9
Pu-241	6.3
Pu-242	33.4
Am-241	2.3
Am-242	0.1
Am-243	17.4
Cm-244	10.8
Cm-245	0.1

Transm. Fuel (final-2)
after 3-y critical, 4-y subcritical

Total	**50**
Np-237	4.4 %
Pu-238	10.3
Pu-239	0.1
Pu-240	7.0
Pu-241	5.0
Pu-242	35.0
Am-241	3.3
Am-242	0.5
Am-243	16.7
Cm-244	16.0
Cm-245 +	1.7

From LWR Waste to Final Waste after Deep Burn Transmutation: mass flow results from the Deep Burn Transmuter studies. The mass flow of the waste transuranics is followed through the Driver and Transmutation Fuel stages of the Deep-burn MHR and a second tier system.

Figure 2-2. DP-MHR Mass Flows

particles in dry and wet repository environments. The predicted failure fraction at one million years was approximately 10^{-4} for wet conditions and 10^{-6} for dry conditions. The results of the calculation are shown in Fig. 2-3. Therefore, this excellent, long-term retention of the products of transmutation accomplishes a reduction in toxicity in the repository equivalent to four to six orders of magnitude of transmutation at the bottom end of the transmutation.

From a toxicity standpoint, irradiation in the DB-MHR, and the isolation provided by the subsequent storage of the corrosion-resistant ceramic-coated particles, lowers the ingested toxicity risk in a repository below that of natural uranium. In Fig. 2-4, the total ingestion toxicity of the direct burial of untreated, spent nuclear fuel is shown in the upper portion of the figure by the curve marked "Total Spent Fuel." The heavy curve in the center of the figure, noted "Total w/Transmutation," shows the reduction of toxicity by transmutation in the DB-MHR and second tier system. The lowest heavy curve, noted "Total w/ Deep Burn Transmutation TRISO barrier," shows the total toxicity available after transmutation and the containment of the residual TRU in the corrosion resistant TRISO coatings. The containment afforded by the coatings, calculated as a fraction of exposed TRU kernels, is taken from Fig. 2-3.

2.2.2.3 Fuel for the Deep-Burn Concept

Ceramic, coated-particle fuel was developed to provide effective radionuclide containment at the high temperatures experienced in MHR cores during normal operation and postulated accidents. A coated particle is a small sphere of the order of 1 millimeter in diameter. The center of the particle is a "kernel" containing the fuel materials in a ceramic form. Coatings that are retentive of fission products surround the kernel. A TRISO-coated particle, to be used in the DB-MHR, is shown schematically in Fig. 2-5; it consists of a spherical kernel containing fissile or fertile material, usually in oxide or carbide form, surrounded by four coating layers: three pyrocarbon (PyC) layers and one silicon carbide (SiC) layer. After early testing confirmed the excellent fission product retention of this design at high temperature, a nearly identical design for the TRISO-coating has been adopted for all international HTGR applications (IAEA-TECHDOC-978 1997).

The diameter of the kernels and the thickness of the buffer layer of the TRISO particle are usually varied depending on the burnup requirement - larger kernel diameters for low burnup and smaller kernels for higher burnup. The typical range of TRISO particle dimensions is shown in Table 2-1.

The TRISO coating system acts as a miniature pressure vessel to contain internal gas pressures generated during fissioning of the kernel material and as a diffusion barrier to contain gaseous and metallic fission products. The coating layer adjacent to the kernel is a low density PyC (buffer layer) containing about 50% voids. This layer absorbs fission product recoils from the kernel, provides a reservoir for fission product gases and carbon monoxide, and accommodates kernel swelling without transmitting large forces to the outer coatings. The next layer is a high-density, isotropic PyC layer that protects the kernel from reactions with chlorine during deposition of the SiC layer, provides structural support for the SiC layer, and protects the SiC from fission products and carbon monoxide during operation.

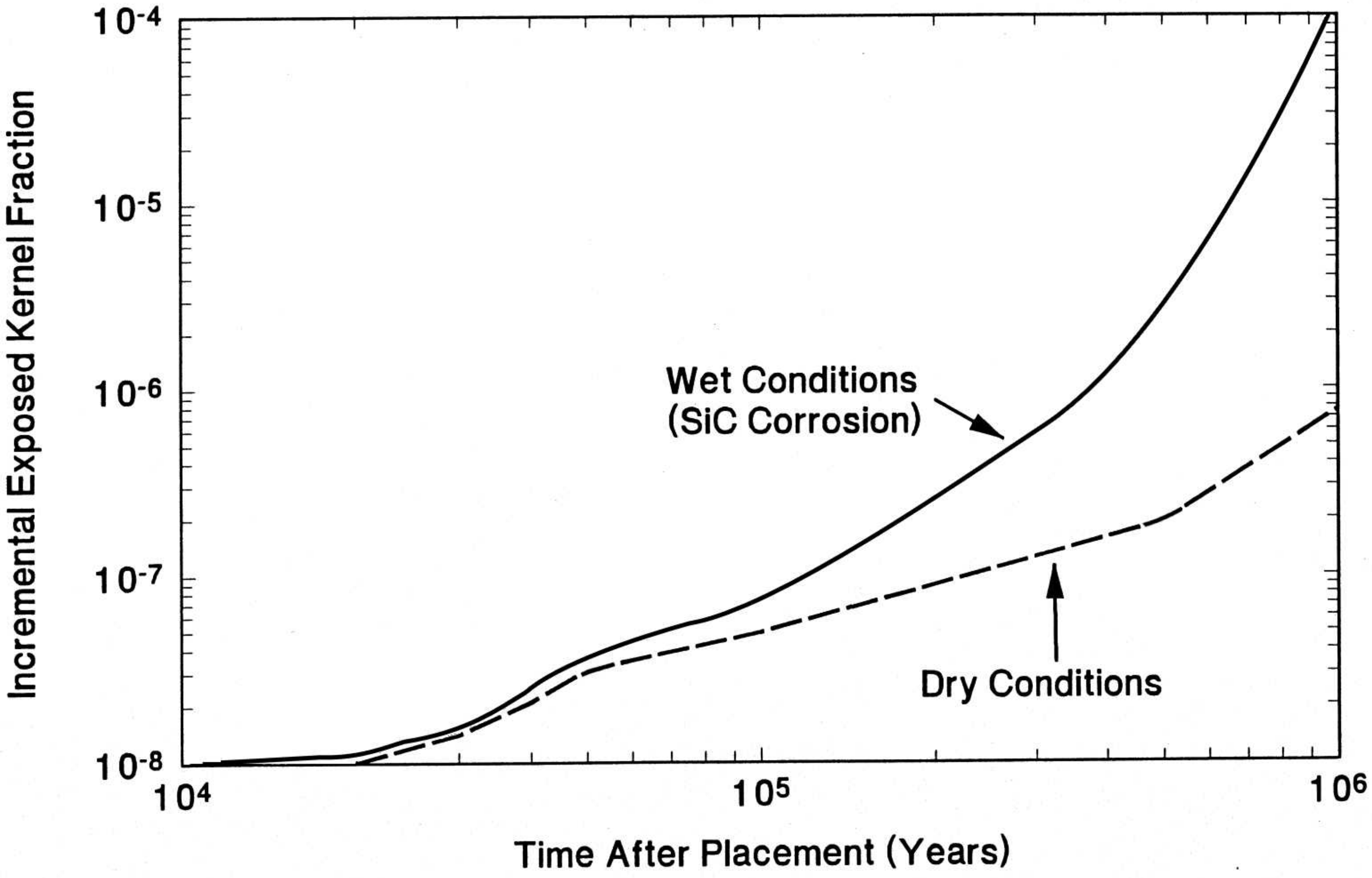


Figure 2-3. Coated Particle Integrity for Long Times in the Repository

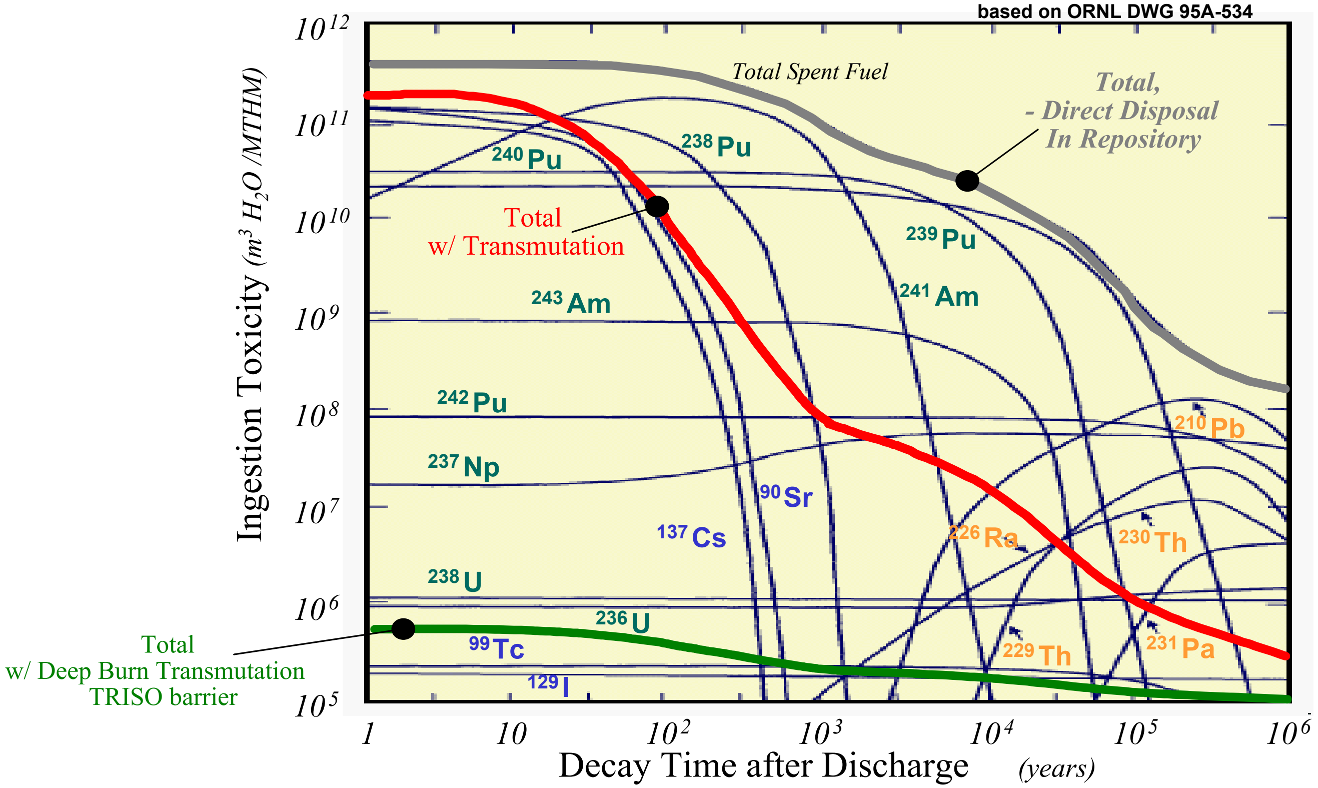
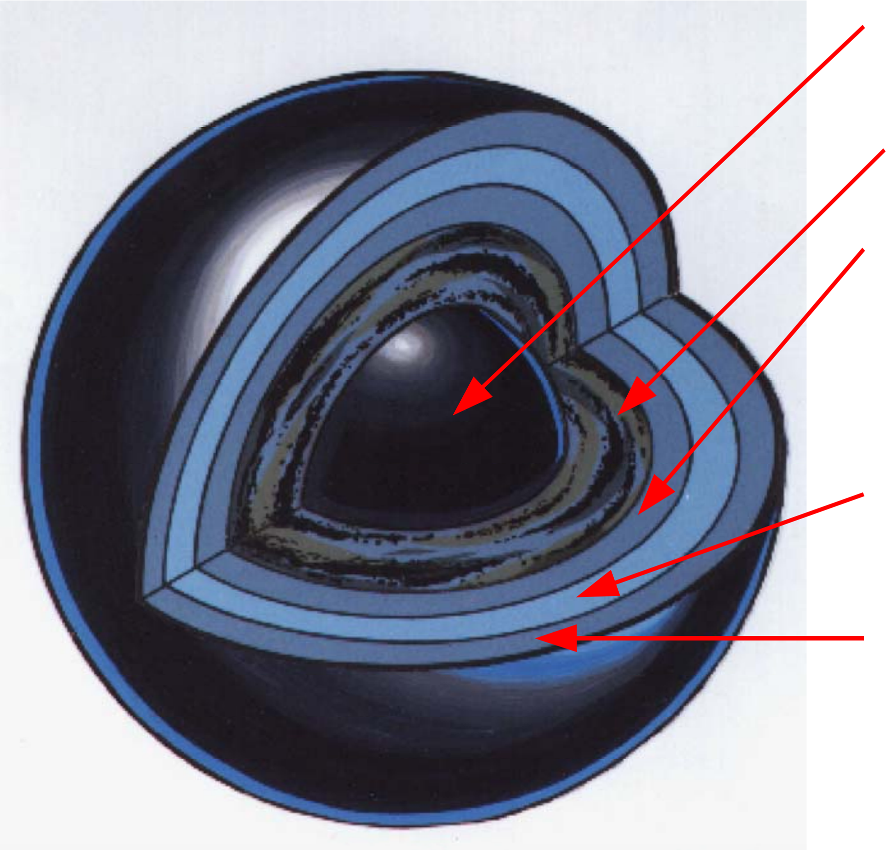


Figure 2-4. Toxicity in Repository for DB-MHR Spent Fuel



COMPONENT/PURPOSE

- Fuel Kernel
  - Provide fission energy/destroy Pu/MA
  - Retain fission products
- Buffer layer (porous carbon layer)
  - Attenuate fission recoils
  - Void volume for fission gases
- Inner Pyrocarbon (IPyC)
  - Provide support for SiC during manufacture and irradiation
  - Prevent Chlorine attack of kernel during deposition of the SiC layer
  - Retain gaseous fission products
- Silicon Carbide (SiC)
  - Primary load bearing member
  - Retain gaseous and metal fission products
- Outer Pyrocarbon (OPyC)
  - Protects SiC from surroundings
  - Hold SiC in compression
  - Retain gaseous fission products

Figure 2-5. Functional Schematic of TRISO-coated Particle

Table 2-1. Typical Attributes of TRISO Particles

LAYER	**THICKNESS (µm)**	**DENSITY (g/cc)**
Kernel	200 - 800	10+
Low density pyrocarbon (buffer)	65 – 100	1.0
Inner high density, isotropic inner pyrocarbon (IPyC)	30 – 40	1.85-1.9
Silicon carbide (SiC)	25 - 50	3.2
Outer high density, isotropic inner pyrocarbon (IPyC)	35 - 40	1.85-1.9

The high-density SiC layer is the strongest layer; it provides the TRISO coating's primary ability to contain the high pressure generated in the kernel/buffer region by fission. The SiC layer is held in compression throughout its irradiation cycle by compressive forces exerted by the dense pyrocarbon layers which shrink under fast neutron irradiation. The outermost layer is another high-density, isotropic PyC layer that protects the SiC and provides structural stability to the particle during irradiation. High Density PyC and SiC layers are impervious to fission gases at normal operating temperatures and the SiC layer is an effective barrier to both gaseous and metal fission products (IAEA-TECDOC-978 1997).

TRISO fuel particles are an ideal form for achieving high neutron efficiency during transmutation (and other missions). They offer the possibility of very high levels of transmutation (Deep Burn) and the diameters and densities of the kernels can be adjusted to preferentially expose non-fissile minor actinides to epithermal neutrons in the TF and fissionable materials in the DF to thermal neutrons. These features are in addition to their ability to sustain high temperatures, which allows production of electricity at high efficiencies and unmatched safety.

The three principal components of the fuel element assemblies for the DB-MHR are shown in Fig. 2-6. The basic fuel unit is the TRISO-coated particle. To complete the fuel assemblies, coated particles are formed into fuel compacts using a carbonaceous thermosetting binder. Compacts are about 12.4 mm in diameter, and 49 mm in length. About 3000 compacts are loaded into blind holes in each graphite fuel element to form a fuel element. The fuel assemblies are hexagonal in cross-section with a height of 800 mm and a cross-flats width of 360 mm. They contain coolant holes parallel to the fuel channels through which helium coolant is passed to remove the heat generated in the fuel.

The prismatic assemblies are stacked into vertical columns (usually 10 assemblies high). The columns are arranged in an annular pattern to form a core with graphite reflector on the inner and outer periphery of the core. Graphite blocks with coolant holes (but no fuel) are placed above and below the fuelled core as reflectors

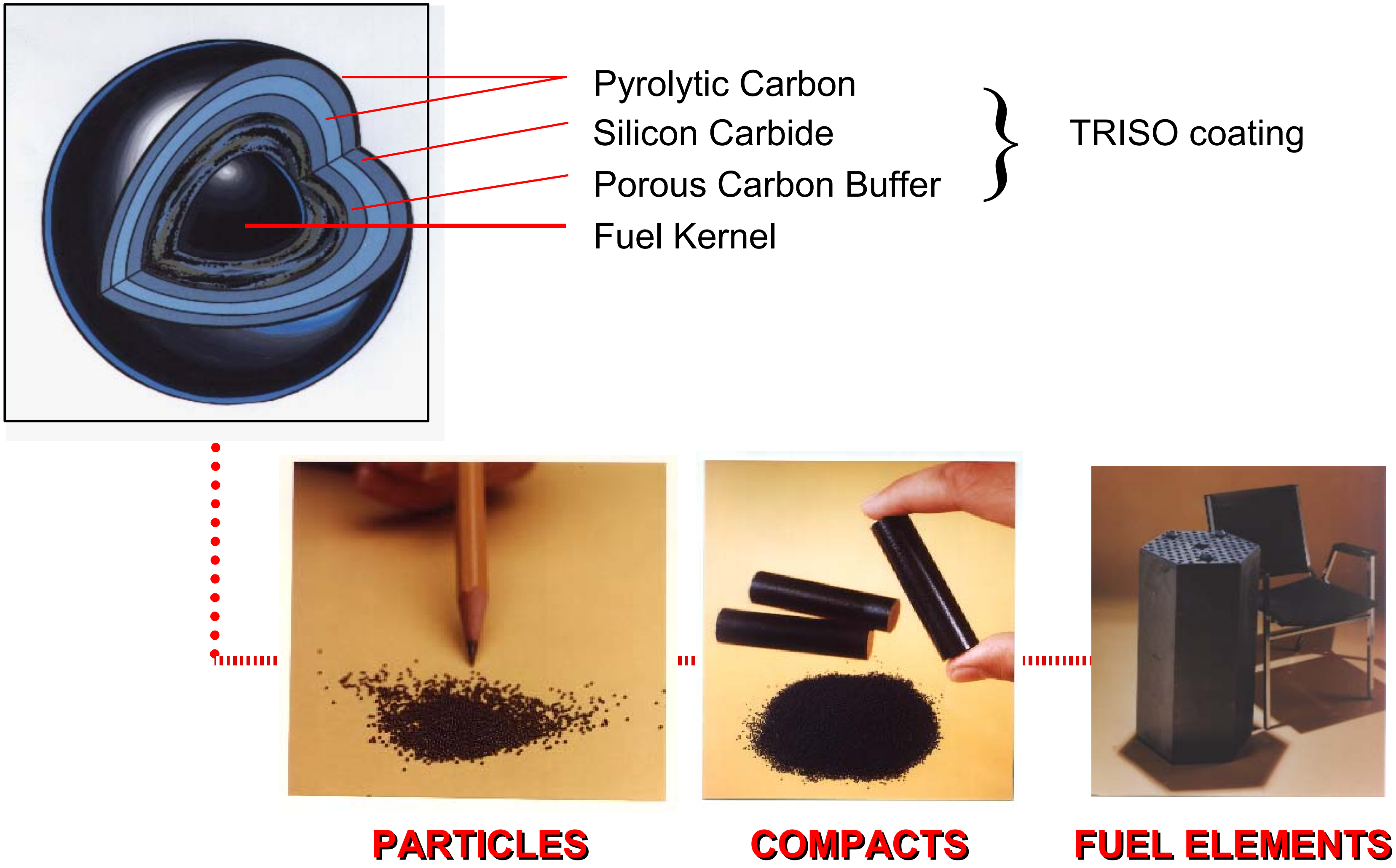


Figure 2-6. DB-MHR Fuel Element Components

The radionuclide containment system for an HTGR is comprised of multiple barriers to limit radionuclide release from the core to the environment to insignificant levels during normal operation and a full spectrum of postulated accidents. The five principal release barriers are (1) the fuel kernel, (2) the particle coatings, particularly the SiC coating, (3) the fuel-element graphite, (4) the primary coolant pressure boundary; and (5) the containment building.

Traditionally, reactor systems using non-fertile fuels (such as in those designed to destroy surplus weapons plutonium) have relied on parasitic burnable poisons (boron, erbium, and others) for burnup and reactivity control roles. In the DB-MHR, the availability of non-fissile transuranics in the TF allows these roles to be taken by the non-fissile transuranics (Venneri 2001). The large n-γ capture cross sections of the non-fissile transuranic actinides contained in the TF cause these actinides to be fertile (that is, they become fissile upon absorption of one or two neutrons). This physical characteristic is advantageous in the DB-MHR in two ways. First, to provide burnable poisons to compensate for the large excess reactivity of the fissile ^{239}Pu and ^{241}Pu; and second, as resonance absorbers to provide prompt negative feedback under all conditions (i.e., an overall negative temperature coefficient of reactivity).

To achieve the Deep Burn design and performance requirements, the DF kernel should have a high density and a large diameter to provide initial self-shielding of some of the fissionable material within the kernel. This self-shielding will reduce beginning of life excess reactivity and allow power to be maintained throughout the planned fuel lifetime, as the fissioning reduces the self-shielding effect of the kernel. In addition, the large diameter protects the inner part of the DF kernel from reactions with resonance region neutrons ($0.15 < E < 0.5$ eV) where ^{239}Pu has an unfavorable capture-to-fission ratio. Non-fissile plutonium and ^{237}Np capture incident neutrons in this energy range in the outer shell of the kernel, leaving the inner part of the kernel with a reduced flux of neutrons in this resonance range. Therefore, more of the fissions in ^{239}Pu are at lower neutron energies where parasitic capture is lower.

The neutrons from the fissioning of DF are used to convert the largely non-fissile isotopes into fissionable isotopes and then to fission them. At the same time the non-fissile nuclides in the TF have resonance capture cross sections at neutron energies slightly higher than the neutron energy characteristic of the moderator temperature. Capture in the TF provides a negative temperature coefficient throughout the cycle. The capture of neutrons in the non-fissionable actinide nuclei therefore performs the useful function of converting the non-fissile nuclei into fissionable nuclei (this is in contrast to the use of burnable poisons, such as erbium, to provide negative temperature coefficient where the neutrons captured do not further the transmutation function). The new fissionable nuclei can then be fissioned, thus destroying long-lived transuranics.

To enhance the transmutation of these largely non-fissile nuclides and to assure the negative temperature coefficient in the TF it is important to enhance the reaction rates - that all of the material in the nuclei in the TF be available to the neutrons, i.e., no self-shielding. This can be accomplished with a small diameter kernel, a kernel diluted with inert material, or a low-density TF kernel.

Developing coated-particle fuel for high burnup and high fast neutron fluences containing undiluted fissile material (as required by the driver fuel) has been carried out with highly-enriched uranium as part of the Fort St. Vrain Nuclear Generating Station Project (FSV) and subsequent commercial reactor development in the U. S. The FTE-13 test in Peach Bottom demonstrated that TRISO-coated undiluted plutonium fuels could be irradiated successfully to high burnup (Sanders 1973 and Miller 1985). Tests of coated plutonium fuel particles were also performed in the Dragon Reactor in the United Kingdom (Formann 1975). The ranges of burnup and fast neutron fluence covered by previous tests are compared to the requirements for the Driver and Transmutation Fuel in Fig. 2-7.

The next step in the development of Driver and Transmutation fuels is to build on this technology to meet the projected fuel requirements. ORNL, with General Atomics, will develop processes and equipment to fabricate DF and TF test samples for irradiation, accident simulation, and other testing.

2.3 DB-MHR Fuel Development Overview

The overall fuel development strategy for the DB-MHR is perhaps best viewed in terms of its four major phases. This development plan covers the first three phases. The fourth phase, not covered here, is the production scale-up, fabrication, and testing of full DF and TF fuel test assemblies for test in a Demonstration Modular Helium Reactor.

All testing will be done in accordance with test specifications, a test plan, and written test procedures. Each full irradiation test of fuel includes pre-irradiation characterization and performance predictions, irradiation, post-irradiation examination, testing under accident conditions, and examination after accident simulation testing. Results will be documented in data compilation reports and interpreted in test evaluation reports.

Phase I of the plan is a Fuel Fabrication Demonstration phase to be completed on or before 2008. During this phase, alternate fuel designs will be tested and the best performing designs selected for further development. TRISO fuel fabrication processes are first re-established and demonstrated using non-fissioning surrogate materials for the kernels. These particles are not irradiated. This work is followed by production of TRISO-coated Driver Fuel particles where HEU UCO has been used as the kernel material instead of Pu/Np. The DF coatings with HEU kernels are then irradiated to high burnup and high fast fluence to test the particle design and that the coatings have been properly fabricated prior to testing Pu/Np fuel fabricated in gloveboxes with plutonium. Tests of alternate DF designs are planned at this point in the development. For example, particles with ZrC coatings replacing the SiC layer or tests of particles with kernels containing features to control the oxygen potential may be tested. After successful irradiation of the HEU/DF particles to study the particle design and the fabrication techniques, actual Driver Fuel is produced and tested.

In the development of Transmutation Fuel (TF), first a HEU analog is also produced and tested. Then small quantities of TF particles are made in a hot cell. Again some alternative designs may be fabricated and tested. Small numbers of these TF particles are made into compacts in a shielded glovebox. The compacts will also contain a quantity of TRISO particles containing fissile material to allow TF to operate at high temperature. The main result of this effort is the selection of the best performing DF and TF designs and processes for further development.

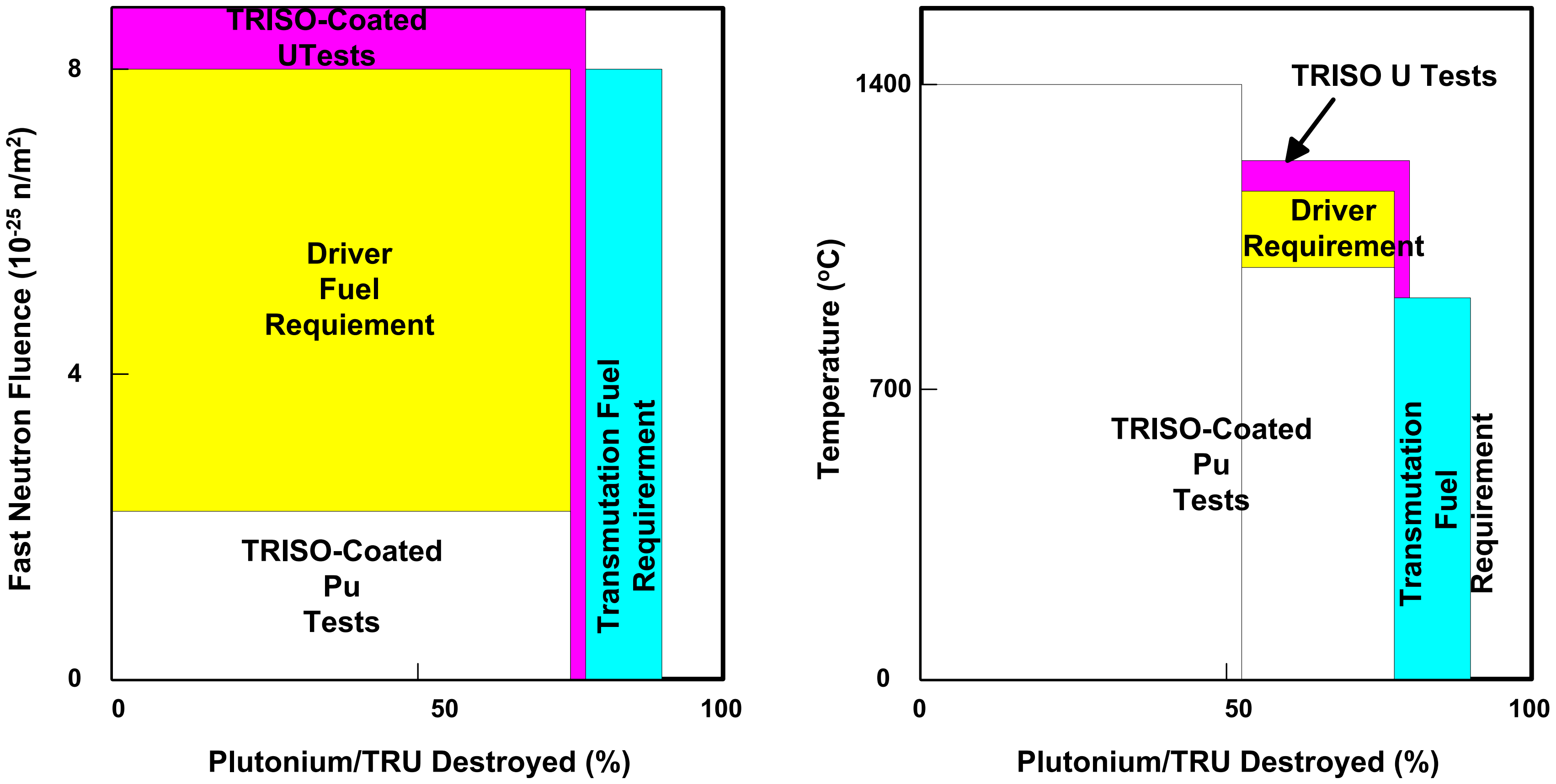


Figure 2-7. Comparison of DF and TF Requirements with the Irradiation

The second major phase (Phase II) of the plan is the proof-of-principle or Qualification phase. DF will be qualified by 2011. In this DF Qualification phase, DF is produced using the reference process developed for the DF in the previous Fuel Fabrication Demonstration phase, irradiated within the envelope of operating and design basis transients, and examined to confirm quality. This phase confirms and qualifies the design and the reference process used for DF fabrication.

In Phase II, TF will also be qualified. The Transmutation Fuel (TF) Qualification phase is similar to the DF Qualification phase discussed in the previous paragraph. The Qualification phase will be completed on or before 2012. In this TF Qualification phase, TF is produced using the reference process developed for the TF in the Fuel Fabrication Demonstration phase, irradiated within the envelope of operating and design basis transients, and examined to confirm quality. This phase confirms and qualifies the design and the reference process for TF fabrication.

The third major phase of the plan consists of a Fuel Validation Tests to be completed in 2012 for DF and by the end of 2014 for TF. In this validation phase, the selected reference DF and TF produced using the designs and the processes qualified for the DF and TF Phase II are tested in statistically significant quantities in multicell capsules. The irradiation conditions will cover the in-core conditions expected in the DB-MHR and will produce irradiated specimens for post-irradiation examination and for accident simulation tests.

After completion of Phase III, DF and TF will be qualified for use in fuel test elements in a DB-MHR and the work of this Fuel Development Plan will be complete. The fuel work will then move to larger scale production and testing of entire fuel elements, fuel segments and full cores of DF and TF in the fourth and final phase. In parallel with the development of DF and TF, the commercial MHR program will design and construct the Demonstration Modular Helium Reactor. This Demonstration MHR will first be operated with commercial LEU TRISO-coated fissile and TRISO-coated natural uranium fertile fuel. After initial satisfactory operation with uranium fuels, test elements containing DF will be introduced into the Demonstration MHR core. When satisfactory operation of DF test fuel has been demonstrated, then test elements containing both DF and TF will be introduced.

Finally, scaleable DF and TF fabrication facilities will be established and the Demonstration MHR core will be converted in stages to all DF/TF fuel for a demonstration of transmutation in the DB-MHR operating mode. This phase will be completed by 2023.

The summary schedule for above fuel development work was presented in Fig. 1-1. Note that this summary representation also shows the relationship between the Deep Burn TRISO fuel development program and the establishment of the Demonstration Modular Helium Reactor.

2.4 Program Organization and Responsibility

General Atomics and the Oak Ridge National Laboratory (ORNL) will carry out the development of coated particle fuel for the DB-MHR. ORNL is the lead laboratory for coated particle transmutation fuel development. General Atomics is the design organization, potential future fuel supplier, and industrial participant in the DB-MHR program.

General Atomics has experience in designing and developing coated particle fuel for Peach Bottom (Turner 1973), Fort St. Vrain (Baxter 1994), large commercial HTGRs (Goodjohn 1982), and Gas-Turbine Modular Helium Reactors (Labar 2002). General Atomics manufactured TRISO-coated uranium fuels for FSV (McEachern/Noren 2001) and for numerous experimental irradiations (IAEA-TECDOC-978 1997). General Atomics and Russian Federation laboratories are currently designing and developing pure plutonium oxide fuels for destruction of excess weapons plutonium (McEachern/Makarov 2001).

ORNL has extensive prior experience developing coated particle fuels, as part of the HTGR and Thorium Recycle programs (Wymer 1968, Lotts 1977, Stinton 1982), and relevant experience fabricating TRU targets containing neptunium, plutonium, americium, and curium as part of the program for production of transuranic isotopes (Chattin 1981).

The roles of the two organizations are defined by a Work Breakdown Structure (WBS), which indicates all of the tasks necessary to complete the fuel development. The descriptions of the specific tasks, and the lead organizations, are shown in Table 2-2.

2.5 Program Coordination and Collaboration

The DB-MHR fuel development program will systematically coordinate its activities with other U.S. and international, coated-particle fuel development activities. Two on-going programs are of particular importance. First, the joint DOE-MD/MINATOM International GT-MHR program (OKBM 1997) for the disposition of surplus Russian weapons plutonium is developing high-burnup, TRISO-coated Pu fuel which is quite similar to the Driver Fuel to be developed by this program. Secondly, the DOE-NE sponsored, AGR fuel development program has been planned to develop and qualify LEU coated-particle fuel to be used in commercial PBMR and GT-MHR designs. Fuel development activities sponsored by the Europe Union and Japan should also produce directly relevant data (e.g., the latter’s planned development of advanced ZrC coatings).

Table 2-2. Responsibility Matrix

WBS	Description	lead
1	Fuel Development Schedule	
1.1	Fuel Fabrication Process Development	ORNL
1.1.1	Base Fuel Fabrication Technology	ORNL
1.1.1.1	Uranium Fabrication Line	ORNL
1.1.1.1.1	Install U Kernel Line	ORNL
1.1.1.1.2	Design/Install Sintering and Coating Line	ORNL
1.1.1.1.3	Design/Install Compacting Line	GA/ORNL
1.1.1.1.4	Install Complete QC	GA/ORNL
1.1.1.2	U-Test Fuel Fabrication	ORNL
1.1.1.2.1	Depleted Uranium Operations	ORNL
1.1.1.2.2	Enriched Uranium Operation	ORNL
1.1.1.2.3	Remote Process Development Support	ORNL
1.1.2.	Driver Fuel	ORNL
1.1.2.1	DF Fabrication Lab (glovebox)	ORNL
1.1.2.1.1	Design/Install DF kernel line	ORNL
1.1.2.1.2	Design/Install DF coater line	ORNL
1.1.2.1.3	Design/Install DF compact equipment	GA/ORNL
1.1.2.1.4	Establish Complete QC for DF	GA/ORNL
1.1.2.2	DF Fabrication	ORNL
1.1.2.2.1	Produce DF kernels	ORNL
1.1.2.2.2	Produce DF TRISO Particles	ORNL
1.1.2.2.3	Produce DF TRISO compacts	ORNL
1.1.2.2.4	Produce DF Margin Test Fuel	ORNL
1.1.2.2.5	Produce DF Validation Fuel	ORNL
1.1.3	Transmutation Fuel	ORNL
1.1.3.1	Remote Fuel Fabrication Line	ORNL
1.1.3.1.1	Design/Install TF resin-loading kernel line	ORNL
1.1.3.1.2	Design/Install TF coater line	ORNL
1.1.3.1.3	Design/Install TF compact equipment	ORNL/GA
1.1.3.1.4	Establish Complete QC for TF	ORNL/GA
1.1.3.2	Transmutation Fuel Fabrication	ORNL
1.1.3.2.1	Produce TF Pu/Am/Cm kernels (cell)	ORNL
1.1.3.2.2	Produce TF Pu/Am/Cm TRISO particles (sparse-glovebox)	ORNL
1.1.3.2.3	Produce TF Pu/Am/Cm TRISO compacts (sparse-glovebox)	ORNL
1.1.3.2.4	Produce TF Margin Test Fuel	ORNL
1.1.3.2.5	Produce TF Validation Fuel	ORNL
1.1.4	Prototypic Scale Refabrication Analyses	GA
1.1.4.1	Preliminary Design of Industrial Scale Refabrication Plants	GA
1.1.4.2	Equipment Analyses for Prototype Refabrication Equipment	GA
1.1.4.3	Economics of Industrial Scale Refabrication Operations	GA

Table 2-2 (continued). Responsibility Matrix

1.2	Fuel Materials Development	ORNL
1.2.1	Driver Fuel Evaluation and Testing	ORNL
1.2.1.1	Non-irradiation Evaluation of Candidates	ORNL
1.2.1.1.1	Prel. DF Thermochemical Analysis	ORNL
1.2.1.1.2	DF ThermoChem Analysis	ORNL
1.2.1.1.3	DF Fuel Compact Thermophysical Properties	ORNL
1.2.1.2	DF Irradiation Tests	ORNL
1.2.1.2.1	Phase 1, Screening of DF Candidates	ORNL
1.2.1.2.1.1	Uranium TRISO Benchmark Tests	ORNL
1.2.1.2.1.2	Uranium Compact Benchmark Tests	ORNL
1.2.1.2.1.3	Zr-SiC DF TRISO Test	ORNL
1.2.1.2.1.4	Zr-Sic DF Compact Test	ORNL
1.2.1.2.2	Phase II, Margin Testing of Reference DF	ORNL
1.2.1.2.2.1	DF Margin Test - Multicell Capsule	ORNL
1.2.1.2.3	Phase III, Fuel Validation Test	ORNL
1.2.1.2.3.1	DF Validation - Multicell Capsule	ORNL
1.2.2	Transmutation Fuel Evaluation and Testing	ORNL
1.2.2.1	Non-Irradiation Evaluation of Candidates	ORNL
1.2.2.1.1	Prel. TF Thermochemical Analysis	ORNL
1.2.2.1.2	TF Thermochemical Performance	ORNL
1.2.1.2.3	TF Fuel Compact Thermophysical Properties	ORNL
1.2.2.2	TF Fuel Irradiation Tests	ORNL
1.2.2.2.1	TF Phase I, Screening of TF Candidates	ORNL
1.2.2.2.1.1	TF1 Compact - Primary Fuel Design	ORNL
1.2.2.2.1.2	TF2 Compact - Backup Design	ORNL
1.2.2.2.2	TF Phase II, Margin Testing of Reference TF	ORNL
1.2.2.2.2.1	TF Margin Tests - Multicell Capsule	ORNL
1.2.2.2.3	TF Phase III, Fuel Validation and Proof Tests	ORNL
1.2.2.2.3.1	TF Validation Multicell Capsule	ORNL
1.3	Fission Product and Actinide Transport, Core	GA
1.3.1	Retention in Fuel Particles	GA
1.3.1.1	Chemical Eval. Of Fission Products in DF and Tf	ORNL
1.3.1.2	Retention in DF and TF Particles	ORNL
1.3.1.2.1	FP-1 Retention in DF Particles - Normal Operation	ORNL
1.3.1.2.2	FP-2 Retention in TF particles - Normal Operation	ORNL
1.3.1.3	Accident Condition Testing of DF and TF Particles	ORNL
1.3.1.3.1	Accident Condition Testing of DF Particles	ORNL
1.3.1.3.2	Accident Condition Testing of TF Particles	ORNL
1.3.2	Retention in Fuel Compacts	GA
1.3.2.1	Retention in DF and TF Compacts	ORNL
1.3.2.1.1	FP-3 Retention in DF Compacts - Normal Operation	ORNL
1.3.2.1.2	FP-4 Retention in TF Compacts - Normal Operation	ORNL
1.3.2.2	Accident Condition Testing of DF and TF Compacts	ORNL
1.3.2.2.1	Accident Condition Testing of DF Compacts	ORNL
1.3.2.2.2	Accident Condition Testing of TF Compacts	ORNL

Table 2-2 (continued). Responsibility Matrix

WBS	Description	lead
1.4	Spent Fuel Disposal Technology	GA
1.4.1	Mechanical Integrity	ORNL
1.4.2	Long Term Performance of Fuel Components	ORNL
1.4.2.1	Fuel Component Performance in Air	ORNL
1.4.2.2	Fuel Component Performance in Groundwater	ORNL
1.4.3	Radionuclide Transport	GA
1.4.3.1	Radionuclide Inventories in Spent Fuel	GA
1.4.3.2	Radionuclide Leaching Rates from Fuel Compacts	ORNL

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3. Fuel Development Status

3.1 Fuel Process Development

Coated particle fuel has been fabricated in many countries throughout the world. It has been used as the fuel in eight power and experimental reactors and has been irradiated in numerous test capsules (IAEA-TEC-DOC-978 1997, IAEA-TECDOC-1198 2001).

3.1.1 Kernel Fabrication

A traditional sol-gel process (developed at ORNL) was used to manufacture the PuO_2 and $(Th,Pu)O_2$ kernels that were irradiated in Peach Bottom fuel test element FTE-13 (Sanders 1973). The substoichiometric oxygen-to-plutonium ratios were obtained by heat treating in pure hydrogen at 1750°C.

BelgoNucleaire manufactured 500-µm PuO_x kernels which were diluted with carbon (C/PuO_2 = ~20) and subsequently TRISO coated for irradiation in the Dragon HTGR and Studsvik Test Reactors (Horsley 1967, Dragon 1968, Bairot 1974). The kernels were prepared by two alternative processes, sol gel and powder agglomeration methods. Considerable emphasis was placed upon finding process conditions (e.g., a high CO overpressure) to prevent Pu volatilization from the kernel during high-temperature sintering and the early stages of TRISO coating.

Ammonia-based precipitation processes developed more recently, with the two most frequently used variants referred to as “internal gelation” and “external gelation,” have rendered the traditional sol-gel processes obsolete. Both internal and external gelation processes use a concentrated heavy-metal nitrate feed that is “acid-deficient”, and therefore require a concentration/denitration solution preparation step. These newer gel-precipitation methods are simpler, more reliable, and more suitable for scale-up and adaptation to radiochemical processing than the traditional sol-gel techniques.

Both internal and external gelation processes have undergone extensive development for the production of microspheres containing UO_2, UCO, ThO_2, and mixed systems of U-Th, U-Pu, and Th-Pu as well as pure carbides and nitrides. An external gelation/precipitation process has been used to manufacture both UCO and UO_2 kernels at GA in the U.S., at Nuclear Fuel Industries in Japan, KFA Juelich, and at HOBEG GmbH in Germany. Kernels of 195-µm diameter HEU UCO were fabricated in laboratory scale equipment at Babcock and Wilcox (now BWXT) by an internal gelation/precipitation process for the NP-MHTGR program. The internal gelation kernels showed more uniform structure and density than similar kernels from external gelation, and on that basis the former process was selected as the reference.

The feasibility of producing 350-µm and 500-µm diameter UCO by internal gelation has been demonstrated, although the quality requirements were not completely demonstrated for UCO material. Only limited work has been done on pure Pu or Pu-rich systems. Plutonium kernel fabrication capability is currently being developed in Russia as part of a U.S. NNSA/MINATOM program to dispose of excess Russian weapons plutonium. A bench-scale Pu fabrication facility is under construction at the Bochvar Institute where high-density, $PuO_{1.7}$ kernels were made in 1996 (McEachern/Makarov 2001). The limited internal gelation trials that have been conducted on feeds containing only plutonium revealed no fundamental barriers, but flowsheet optimization will be needed to avoid conditions that lead to cracking of the microspheres (Forthmann 1977). No significant complications are expected from inclusion of 5% Np in this process.

GA and ORNL also developed a weak-acid resin (WAR) process for the production of UC_xO_y and UC_2 kernels (Dobson 1980). The WAR process was suitable for hot cell use for fabricating kernels from ^{233}U recovered in the uranium/thorium fuel cycle which was the reference fuel cycle for early U.S. HTGRs. In this process Uranyl ions (UO_2^{++}) were loaded on to a weak-acid resin in its ammonium form and subsequently dried, calcined, and converted to the desired stoichiometry. The resulting product was a porous ~300-μm kernel which proved to have a fission gas release rate several times higher than fully dense kernels. Because of this high gas release and heavy-metal loading limitations, development of WAR kernels was terminated in 1977.

A traditional sol-gel process for preparing curium oxide for transmutation targets for heavy element production was developed and used at ORNL from 1968 through 1970, but this method was plagued with erratic operation and poor yields (Burch 1970). Subsequently, a simpler and more reliable strong-acid resin process was developed at ORNL for making Am-Cm oxide for transmutation targets in an aluminum based CERMET (Chattin 1981). This process has been in operation for over thirty years at a lab-scale, and has been proven suitable for remote operations. Modifications of this process to make larger particles of a different chemistry would be necessary to support the fabrication of TF.

3.1.2 Coated Particle Fabrication

TRISO coatings have been applied to various fuel kernels, including PuO_2, $(Th,Pu)O_2$, PuO_x/C, NpO_2, UO_2, UCO, UC_2, $(Th,U)C_2$, $(Th,U)O_2$, ThC_2, and ThO_2 kernels, in the U.S., Europe, Japan and China.

The basic principles for applying TRISO coatings on dense microspheres by chemical vapor deposition are well established after nearly four decades of international development. The exhaustive study by Voice in the 1960s of the relationships between coating process parameters and the attendant physical properties of the SiC coating remains a seminal work (Voice 1969). Additional TRISO coating process experience in Germany and the United Kingdom has also been described (Huschka 1977, Lefevre 1977). Nevertheless, in certain regards, the production of high-quality coatings, especially PyC coatings, is as much art as science. There is not yet a set of measurable PyC properties that can insure good irradiation performance of the pyrocarbons with high confidence. To assure good irradiation performance, a combination of process equipment, process procedures, and measurable properties must be used.

There is a small amount of experience with pure plutonium oxide kernels that are similar to the Driver Fuel. The PuO_2 and (mixed ThO_2, $PuO_{1.7}$) kernels irradiated in Peach Bottom fuel test element FTE-13 were TRISO-coated in a laboratory scale coater at ORNL (Sanders 1973). The pyrocarbon coatings were conventional, in that the buffer layer was derived from acetylene and the low-temperature isotropic (LTI) IPyC and OPyC coatings were derived from propylene.

However, the reported active coating gas concentration for the PyC was > 50% of the total gas content, much higher than now used to produce good coatings. These PyC layers were coated at temperatures between 1375°C and 1400°C, somewhat higher temperatures than used today . The SiC coatings were deposited at 1500 to 1550°C from methyltrichlorosilane with a hydrogen carrier gas.

The PuO_2 and PuO_2/C kernels irradiated in the Dragon HTGR and at Studsvik were TRISO-coated in a small laboratory coater. The coatings were of an unusual design. Only a thin 10 μm buffer and a 1 to 2 μm porous PyC transition layer was placed on either side of a 50 to 60 μm SiC layer. The PyC coatings were deposited from an acetylene/hydrogen/CO mixture at 1500°C. Moreover, a very low coating gas concentration was used; these conditions are expected to produce highly oriented coatings. The SiC coatings were deposited at 1450 to 1500°C from methyltrichlorosilane with a hydrogen carrier gas (Horsley 1967). Different coating conditions were investigated with the objective of minimizing potential plutonium volatilization from the kernel during coating.

Conventional TRISO coatings have been successfully applied to small quantities of $^{237}NpO_2$ kernels in a laboratory-scale coater at General Atomics. TRISO coatings have (apparently) not been applied to kernels with a high americium content.

General Atomics produced large quantities of TRISO-coated HEU $(Th,U)C_2/ThC_2$ fuel (33,000 kg HM) for the Fort St. Vrain HTGR (McEachern/Noren 2001). All of the FSV fuel met the applicable products specifications; however, the as-manufactured heavy metal contamination and SiC coating defect fractions allowed for FSV fuel were at least an order-of-magnitude higher than that required for the DB-MHR. The German company, HOBEG (owned by NUKEM and GA), mass produced TRISO-coated LEU UO_2 reload fuel for the AVR that met the as-manufactured quality requirements tentatively adopted for the DB-MHR reactor and performed very well at low burnup and low fast neutron fluences.

3.1.3 Compact Fabrication

Extensive experience in the production of cylindrical fuel compacts has been gained at GA during production of FSV fuel (McEachern/Noren 2001) and through later process development for the MHTGR (Goodin 1996). Similar experience has been demonstrated for spherical fuel elements with overcoated fuel particles for the pebble bed reactor programs in Germany (Nabielek 1990) and China (Chunhe 2001). Annular fuel compacts using a similar overcoating process have been developed in Japan (Saito 1991).

Two different processes were used to fabricate the fuel compacts containing the TRISO-coated PuO_2 and $(Th,Pu)O_2$ particles irradiated in Peach Bottom fuel test element FTE-13 (Sanders 1973). Most of the fuel compacts were made with a slug-injection process, but the compacts containing the smallest diameter particles were fabricated by an admix process. FSV compacts were fabricated early with coal-tar-pitch matrix and later with petroleum pitch matrix. The FSV compact matrix was "thermo-plastic." This means the matrix softened during the carbonization step (at about 800°C) and required support in a bed of alumina powder to maintain the required dimensions. All compacts were cleaned with HCl and given a final heat treatment in argon at temperatures up to 1800°C.

The product specifications for FSV fuel were much less restrictive for heavy-metal contamination and SiC defects in completed fuel compacts than the provisional requirements for the DB-MHR core. The compacting process for DB-MHR fuel must be improved to reduce the level of heavy-metal contamination and defective particles in compacts by more than an order of magnitude compared to the levels demonstrated during FSV mass production.

A compacting process improvement program was conducted at General Atomics in 1995-1996 (Goodin 1996). High-quality, German LEU UO_2 particles were used in the study to demonstrate fuel meeting commercial GT-MHR compact fuel quality specifications for as-manufactured heavy-metal contamination and coating defects in the finished compacts (the quality requirements for the DF compacts are essentially the same). Improvements were made in the control of the bed compacting and matrix-injection steps, which significantly reduced the mechanical stresses in the coated particles during the formation of green compacts.

More importantly, these process studies confirmed the results of earlier studies: that a major source of heavy-metal contamination and SiC coating defects was from chemical impurities, especially transition metals, in the petroleum-pitch matrix and introduced into the compacts in the heat treatment furnace. Iron was the most active contaminant, which attacked the SiC coating during the high-temperature heat treatment step. Process changes were made to reduce the impurity levels by scavenging metals from the petroleum pitch, and the furnace internals were efficiently cleaned with HCl prior to firing. As a result of these improvements, fuel compacts (referred to as "MHR-1 compacts") were fabricated with conventional TRISO particles that met commercial GT-MHR product quality specifications with large margins.

For Driver- and Transmutation Fuel compacts it is desired to use a compact process more adaptable to glovebox and remote operation and one that generates less contaminated waste than the thermoplastic, petroleum-pitch process used for FSV fuel. The new process involves the use of powdered graphite impregnated with a man-made thermosetting resin. This matrix composition is similar to the one used for the spherical fuel elements developed by HOBEG (Heit 1985). An injectable thermosetting resin process has been studied for fabrication of compacts (Bullock 1976, Bullock 1977, Bullock 1980, Bullock 1981). The French company CERCA studied a process where matrix flakes were mixed with particles and the compacts formed in a mold by melting the binder; apparently with more success. It is the latter process that is to be developed for DF and TF compacts.

3.1.4 Quality Control Techniques

The QC techniques currently available for inspection and testing of DB-MHR fuel components are essentially the methods used for inspection of FSV fuel and improved for GT-MHR fuel (Bresnick 1991). Tests suitable for uranium fuels exist for all of the fuel attributes and parameters now specified for uranium fuels. For use with Driver Fuel and Transmutation Fuel the uranium methods must be adapted for use in glove boxes and some in hot cells. There are appropriate chemical analyses available for determining quantities of plutonium with the required accuracy and precision. Because both DF and TF contain mostly plutonium, important measures such as burn-leach determination of SiC layer defects and compact contamination can be used but based on measurements of plutonium released. If sufficiently small samples can be used, it may be possible to carry out many of the tests on TF in shielded glove boxes rather than in the hot cell.

Because the SiC layer is the primary fission product barrier it is important to have methods to measure its integrity and its structure. A SiC defect fraction specification has been used historically to control the defect level in as-manufactured SiC coatings. The method that has been universally used to measure the defective SiC coating fraction is the burn-leach test. The burn-leach test will be adapted to the glovebox and hot cell environment and to the new kernel compositions. Other methods for SiC defect detection will be explored in the QC improvement task.

The poor performance of the fuel in capsule HRB-21 and in the NPR capsules was due primarily to SiC failure caused by a poor coating design and excessive irradiation induced anisotropic shrinkage of poorly made IPyC coating (Leikind 1993). Structural isotropy is known to be a key material property correlated with PyC dimensional stability under irradiation, but optical anisotropy (BAF_o), the technique currently used QC technique, does not correlate well with OPyC failure data. Furthermore, the BAF_o values measured for the IPyC coatings in the fuel irradiated in HRB-21 and in the NPR capsules were well within the specified limits. Thus, the acceptable range of BAF_o, as specified in current uranium fuel specifications, appears to be an inadequate indicator of the both IPyC and OPyC coating performance under irradiation. In addition to the BAF_o, other promising methods to characterize the pyrocarbons have been explored but none have been studied sufficiently to allow them to be used as routine QC methods (Bokros 1966, Pollmann 1977, Krautwasser 1983).

In the absence of fully qualified techniques for characterizing PyC properties, process specifications are an effective method of assuring adequate PyC irradiation performance. Consequently, Strict process control and process specifications will be used while improved PyC characterization techniques are being developed as part of the QC development program.

The IPyC coating must be sufficiently impermeable at the SiC coating temperature (≥ 150°C) to prevent HCl gas, a by-product of the thermal decomposition of methyltrichlorosilane, from reaching and reacting with the kernel during application of the SiC coating.

Historically, the permeability of the IPyC coating layer has been “controlled” by the IPyC coating deposition conditions and the coating density specification under the assumption that a sufficiently high IPyC density ensures low permeability. The QC method historically used to measure the IPyC coating density is the liquid gradient column (or “sink float”) method. This method is quite reliable except for coatings with high open porosity where the gradient column fluid fills the open pores, giving an erroneously high measured density - leading to non-rejection of a porous coating. The allowable permeability of the IPyC depends in part on the reactivity of the kernel with HCl.

If the IPyC coating is not sufficiently impermeable, HCl gas may react with the kernel during deposition of the SiC coating to: (a) form volatile compounds that can cause defects in the SiC layers to form during coating, or (b) form volatile uranium chlorides that can diffuse into the buffer and IPyC coatings dispersing the uranium into the coating. This latter condition, called "fuel dispersion," increases the probability of SiC failure due to fission product attack during irradiation. Fuel dispersion mechanisms also exist for kernels containing Pu and Am. The effectiveness of the IPyC in protecting the kernel is "controlled" by the fuel dispersion specification applicable to the fuel particles in fired fuel compacts. However, it is not clear that the fuel dispersion measurement is sufficiently sensitive to detect low levels of dispersion that could be significant with respect to the formation of SiC defects during irradiation. Studies to define the permeability of the DF and TF coatings, which can achieve the required irradiation performance, are part of the fuel specification and process development activities included in this plan. IPyC permeability and isotropy will be investigated as part of the coating process development activities.

SiC coatings with excessive free carbon in the grain boundaries, poor grain structure, internal defects such as soot or impurity inclusions or stacking faults, surface cracks, excessive surface roughness, or other unfavorable properties may cause the SiC coating to be prone to fission product leakage or structural failure during irradiation. From a QC standpoint, the SiC defects that are controlled by the defective SiC specification and that are measured by the burn-leach test are defects that penetrate the SiC coating. Advanced QC techniques which promise to be more sensitive for detecting SiC defects than the conventional burn-leach method will be investigated.

The irradiation performance of TRISO-coated plutonium oxide fuel is known to be sensitive to the O/Pu ratio that is controlled by the product specification. Qualified methods are readily available for measuring the O/Pu ratio in bulk samples. A number of candidate method for determining the O/Pu ratio on individual coatings appears feasible using available equipment but a procedure for these measurements must be worked out.

While most of the QC methods are adequate for direct use or are adaptable to use for DF and TF, some work on the techniques mentioned in the previous paragraphs could profit by some exploration of improvements in this method.

In addition, many of the available methods employ technologies that are inherently time consuming and labor intensive. Although adequate for inspection of relatively small quantities of fuel in support of the fuel development activities described in this plan, these techniques would likely be too costly to use in a large-scale fuel production facility. Some of the methods have to be adapted to glovebox operations for DF and to hot cell operations for TF fuel manufacturing. There are tasks in this plan to study the automation of the QC techniques where feasible to reduce costs and improve accuracy.

3.1.5 Product Recycle and Waste Treatment

The recovery of uranium from scrap materials was demonstrated in the U.S. during the manufacture of Fort St. Vrain fuel (33,000-kg heavy metal), when process units were developed to recover highly enriched uranium on a continuous basis (Bresnick 1991). Of particular note, the experience applicable to the crushing and dissolving of out-of-specification TRISO-coated particles to recover the heavy metal is extensive, including data from fuel reprocessing developmental tests under a previous DOE program in the 1970s (Young 1977, Carney 1979, GA-A17700 1984).

The U.S. nuclear weapons programs and previous civilian nuclear reprocessing programs have developed processes and techniques for recovering transuranic nuclides from a wide variety of process streams for recycle and/or disposal. Much of this process technology should be directly applicable to the processes which will be used to fabricate Driver and Transmutation Fuel for the DB-MHR, including the liquid waste streams generated during kernel manufacturing and the gaseous waste streams generated during coating.

Each of the steps for fuel fabrication, from kernels through fuel compacts, will generate quantities of liquid and solid waste products that must be recycled or disposed of. In addition, a number of the waste streams will contain mixed wastes that need special attention.

3.2 Fuel Materials Development

3.2.1 TRISO-Coated Fuels in National Programs

Coated fuel particle and compact designs have been the subject of development for more than four decades. Over this period the TRISO particle fuel has been tested or used as the primary fuel in three power reactors and five test reactors. These reactors and their fuel are described in Table 3-1. Additional programs are currently underway in Russia and South Africa to utilize the TRISO particle design in those nations (McEachern Sep/2001). The successful production of over 35,000 kg of TRISO fuel in support of these reactors was based on technology developed within each nation or obtained from cooperation by international contributors.

The cooperation and sharing of information has led to a similarity in TRISO coating designs for reactors in different countries. This point is illustrated by the comparison of TRISO designs in each nation as shown in Table 3-2.

The wide international development and the use of the common TRISO fuel designs have provided a solid technology base. That technology base supports the application of TRISO fuel in the transmutation of radioactive by-products from nuclear energy production

In the evolution of the TRISO particle designs, there has been extensive international work on development of materials for improved fission product retention performance. Much like the TRISO design comparison for reactors in different nations, a striking similarity in specified material properties has been developed as a result of experience within individual nations and shared technology. The following discussion provides a summary of the status of international TRISO fuel materials development.

Table 3-1 Reactors Using Coated Particle Fuels

REACTOR	TYPE	COUNTRY	FUEL TYPE	OPERATING TIME	POWER (MW_t)
Dragon	Test	United Kingdom	Various (including TRISO)	1964-1977	20
Peach Bottom I	Prototype Power	United States	$(Th/U)C_2$ BISO (test elements)	1966-1974	115
UHTREX	Experimental	United States	UC_2 Triplex	1967-1970	3
AVR	Test	Germany	Various (including TRISO)	1966-1989	46
FSV	Power	United States	$(Th,U)C_2$ TRISO ThC_2 TRISO	1976-1989	842
THTR	Prototype Power	Germany	$(Th/U)O_2$ BISO	1983-1989	750
HTTR	Test	Japan	UO_2 TRISO	1998-	30
HTR-10	Test	China	UO_2 TRISO	2000-	10

Table 3-2. Comparison of TRISO Particle Designs

ITEM	FRG	U.S.		JAPAN	USSR	CHINA
Kernel		Fissile	Fertile			
Composition	UO_2	UCO	UCO or UO_2	UO_2	UO_2	UO_2
Diameter (μm)	500	350	500	600	500	500
Enrichment (%)	10.6	19.9	natural	3.4-9.9	8.0(2)	<20
Heavy metal loading per compact or sphere (g)	7	Varies	varies	13.5	7.0(2)	5
Coating thickness (μm)						
Buffer	95	100	65	60	90-100	90
Inner PyC	40	35	35	30	70-80	40
Silicon carbide	35	35	35	25(1)	60	35
Outer PyC	40	40	40	45	60	40
Protective overcoating (μm)	~200	None	none	~200	-	~200
Fuel element						
Type	Sphere	Block		Block	Sphere	Sphere
Graphite grade	A3-3	H-451		IG-110	30PG MPG-6	Chinese grade
No. of coated particles per compact or sphere	~11,000	~6,500	~3,700	~13,500	13,000-16,000(2)	8,300
Volume packing fraction (%)	10	51		30	12-15(2)	4
Design limits						
Temperature (ºC)	1250	700-1250		1495	1400(2)	1000
Burnup (% FIMA)	8-10	≤26	≤7	3.6	8.8(2)	11.0
Neutron fluence (10^{25} m^{-2} E>16 fJ)	2-3	4.5 (E>29 fJ)		1.3 (E>29 fJ)	3.0(2)	1.0

(1) Specification value for HTTR initial core, as-built value is 25-35 μm

(2) Specification value for VGM reactor

United Kingdom: The earliest work on TRISO took place in the Dragon Reactor Program in the United Kingdom. Particles were made with spherical carbide and oxide fuel kernels of 200 to 1000 μm diameter, with controlled porosity of 5 to 20% of the kernel volume. This free volume accommodated fission gases and carbon reaction gases, and kept the total gas pressure at acceptably low levels. The Dragon Program was the first to use SiC coatings on gas-cooled reactor fuel particles and Eric Voice thoroughly explored the relationship between SiC structure and deposition conditions (Voice 1969). The Dragon Program identified the importance of controlling the pyrocarbon (PyC) and silicon carbide (SiC) Chemical Vapor Deposition (CVD) process parameters to achieve product qualities for improved coating performance during irradiation. The TRISO fuel was fabricated with a defect fraction of less than 10^{-5} by establishing process specifications to minimize coating-kernel bonding and misshapen kernels (IAEA-TECDOC-978 1997).

One of the Dragon Program contributions most relevant to the DB-MHR Fuel Program is the demonstration that TRISO coated plutonium oxide fuel retained excellent coating integrity at high burn ups and at temperatures higher than expected for the DB-MHR (Formann 1975). Because of a national selection of a competing power generation source, the Dragon Program TRISO fuel development effort was terminated in 1974.

Germany: The earliest work in the Federal Republic of Germany (FRG) was to develop high-enriched (Th, U)O_2 particles coated with two layers of isotropic PyC, and this coating system was called BISO. The BISO particles were molded in a matrix within a high strength 60 mm diameter warm-pressed and heat-treated graphite sphere known as a "kugel" or pebble. This BISO fuel form was licensed for use in the AVR and THTR reactors in the 1970s.

The need for improved fission product retention provided by the SiC coating of the TRISO design was recognized, and the German development effort led to adoption of TRISO coated low enriched uranium (LEU) UO_2 fuel as the reference fuel in the early 1980s. The German program was successful in achieving fuel quality with a mean defective coating fraction of $<3\text{x}10^{-5}$. That low value corresponded to about one defective particle for every 2 to 3 of the pebble fuel elements. Each pebble contained about 11,000 TRISO particles. Irradiation tests showed that non-defective particles did not fail during irradiation. The TRISO-coated UO_2 particles were irradiated to up to 11% burnup and fast neutron fluence up to $6\text{x}10^{25}$ n/m^2, E>16fJ at temperatures up to 1150°C. A few German irradiations were carried to 15% burnup to fluences as high as $8\text{x}10^{25}$ n/m^2 (E>16 FJ) at temperatures below 1075°C. Other German fuel samples were irradiated to 15.3% burnup and fast neutron fluence of about $2\text{x}10^{25}$ n/m^2 at temperatures up to 1320°C (IAEA-TECDOC-978 1997).

The German test program also subjected irradiated fuel with up to 11% FIMA to simulated accident conditions that demonstrated no particle failure after 500 hours at 1600°C (IAEA-TECDOC-978 1997). Fuel with 14% FIMA showed onset of failure in a shorter time of 100 hours at 1600°C. These results indicate the importance of characterizing accident condition performance in fuels over the full design and expected burnup range. Because of commercial decisions to shut down the THTR and to delay the deployment of nuclear power plants in Germany, the HOBEG Company decided to abandon production and development activities in 1989. Specialists in German TRISO fuel technology remain active at Forschungszentrum

Juelich (FZJ) and continue to contribute to exchange of fuel process, design, and performance information at the international level.

United States: The technology for fabrication of TRISO-coated plutonium oxide similar to the DF was developed on a laboratory scale at ORNL in the 1970s. Parametric studies were conducted with experimental batches of TRISO-coated fuel containing a range of oxygen/ plutonium (O/Pu) ratios. The purpose of this work was to determine if reduction of internal CO gas pressure and improved performance could be achieved by use of kernels made up of mixtures of PuO_2 and Pu_2O_3. Those fuel particles were irradiated in the Peach Bottom HTGR to the burnup now required for the DB-MHR first tier fuels. Kernels with an O/Pu ratio of 1.7 were found to exhibit less kernel migration and superior irradiation performance relative to kernels with higher O/Pu ratios (Miller 1985). This early work with TRISO plutonium oxide fuel is directly applicable to the DB-MHR TRISO Fuel Development Program and provided guidance for work described in this program plan. The work with Pu based TRISO fuel in the U.S. was preceded and followed by extensive experience with uranium and thorium based fuel particles.

Similar to the German experience, the earliest U.S. coated particle fuels were the BISO type manufactured in the early 1960s for the Peach Bottom HTGR. The TRISO coating was adopted in the late 1960s to provide superior fission product retention in the FSV HTGR. General Atomics manufactured the fuel in the first large-scale production of TRISO coated particles (McEachern/Noren 2001). The FSV reactor used $(\sim4Th,U)C_2$ fissile and ThC_2 fertile fuel. The fuel performed well at operating temperatures over 1200°C, attaining a fissile particle burnup of 16% FIMA and a fast neutron fluence of $4.5x10^{25}$ n/m^2 (E>29 fJ) [design values were 16% FIMA and 8 x 10^{25} n/m^2, respectively]. Fission products were retained in the fuel as evidenced by the low radionuclide content of the circulating helium coolant and low levels of radionuclides plated out on the primary circuit (Baxter 1994).

Early success with manufacture and irradiation of HEU UC_2 TRISO particles in the 1970s prompted the licensing of that fuel for use in FSV reload segments (GA 1978). Key technical bases for granting of the license by the NRC were results from the P13R and P13S capsule irradiation tests. In those tests, HEU UC_2 TRISO was taken to 75% burnup at 1050°C. In samples where the outer OPyC was not sufficiently isotropic, the OPyC had high failure and 3 to 6% of the SiC coatings failed. However, in particles that met specification requirements, where the OPyC survived irradiation, the coating failure was 10^{-3} fraction or less (Scott 1976, Harmon 1977).

Beginning in the 1970s, General Atomics designed large [800 - 1000 MW(e)] commercial power reactors and offered them to utilities. After use of HEU in commercial reactors was stopped, the modular helium reactor (MHR) was developed and the fuel for the MHR concept had a 350 μm diameter LEU (19.9% enriched) fissile kernel, consisting initially of UC_2, and later a uranium oxycarbide (UO_2 and UC_2 mixture) referred to as UCO. Extensive work was done on commercial fuel processes and on performance tests of various uranium/thorium and LEU fuel cycles.

Following FSV fuel development, and starting in the 1980s, there was a New Production Reactor (NPR) design (for tritium production) and a series of MHR designs considered for commercial power applications in the U.S. The fuel for the MHR concept had a 350 μm diameter LEU (19.9% enriched) fissile kernel, consisting a uranium oxycarbide (UO_2 and UC_2 mixture), traditionally referred to as UCO. The NPR fuel kernel was HEU UCO, of 200 μm diameter. The MHR fertile fuel kernel was initially ThO_2, but later selected to be natural uranium UCO, of 500 μm diameter. Both the fissile and fertile fuels were TRISO coated. The maximum exposure conditions required for the MHR were 26% FIMA fissile, 7% FIMA fertile, and fast fluence of 5×10^{25}, E>29fJ, with a maximum operating temperature of 1250°C and a maximum accident temperature of 1600°C (McEachern 2001). The NPR exposure conditions were 75% FIMA burnup and fast fluence of 5×10^{25} n/m^2, with a maximum operating temperature of 1250°C and maximum accident temperature of 1600°C.

Multiple irradiation test capsule experiments were performed as part of the MHR and NPR TRISO fuel development programs. Early in the test program, damage to fuel particles during the compact fabrication process resulted in exposed uranium of up to 10^{-4} fraction of total uranium. In an attempt to reduce damage to particles during compact fabrication, a design modification was tested in the early 1990s where a sacrificial porous PyC layer was deposited over the OPyC to protect the particle during compact fabrication. The IPyC was make thicker (and unfortunately more anisotropic) and less porous than the normal TRISO coating to prevent HCl reactions with the kernel and concomitant formation of SiC defects during coating. This coating design was called TRISO-P after the sacrificial protective layer. The protective layer resulted in less particle damage, and the uranium contamination level of compacts was reduced to less than 10^{-5} fraction.

Four irradiation capsule experiments (HRB-21, NPR-1, NPR-1A and NPR-2) were carried out with the new experimental coating design. The HRB-21 fuel contained LEU UCO fissile kernels and natural U fertile kernels while the NPR capsules contained HEU UCO kernels. The reduced contamination caused the initial fission gas release to be about two orders of magnitude less than for fuel in previous test capsules.

The design changes introduced into the TRISO-P coating lead to the high levels of coating failure in these capsules. The failures resulted from poor pyrocarbon coating performance. The coatings in these four capsules failed at nearly the same fast neutron fluence ($\sim 2 \times 10^{25}$ n/m^2) regardless of the burnup in the particle. At increasing neutron exposure there was higher than normal failure of the OPyC coatings with OPyC failure of essentially 100% in the high exposure, high temperature compacts. The thicker, more anisotropic IPyC coatings also failed in greater numbers than expected in the high fast fluence, high temperature compacts. SiC failures were noted to be associated with IPyC cracks.

Later analyses (Leikind 1993) indicated that the SiC failures were caused by high local stresses induced at the inner surface of the SiC by the fast neutron induced shrinking in the attached, cracked IPyC. In addition stresses in the SiC stresses were increased by the failure of the OPyC layer. The fission gas release rate increased to levels similar to prior fuel designs. Metallographic examination of compacts after irradiation showed up to 6% fraction SiC failure in compacts where OPyC failure had been extensive. These results were similar to those observed previously for high burnup HEU UC_2 TRISO fuel with high OPyC failure. (Scott 1976, Harmon 1977). This performance may be compared to results from more successful irradiations such as GF-4, SSL-2, R2-K13, HRB-15A, P-13R and P-13S capsules (IAEA-TECDOC-978 1997).

DF and TF coating designs are of the standard TRISO type so difficulties arising from the TRISO-P features will not affect performance. Fuel coating development proposed in this program is expected to reduce that failure fraction further, to meet a DB-MHR requirement of $\leq 10^{-4}$ fraction failed at peak burnup. In addition, comparison of U.S. and German TRISO coating properties suggests that performance improvements in the U.S. fuel may be achieved by process adjustments, yielding more complete PyC isotropy and finer grain size in the SiC layers (Petti 2002).

Subsequent work at General Atomics has shown that, by careful control of impurities in the matrix material, bed compaction forces, injection parameters, and impurities in the high temperature furnace that particle damage can be avoided in compacts fabricated using particles of the standard TRISO design (Goodin 1996).

A lack of commercial interest in the construction of reactor power plants has caused a contraction in availability of coated particle fabrication facilities. During 2002, lab-scale facilities for production of non-radioactive and uranium TRISO-coated particles have been reestablished at ORNL, and kernel and coating process studies are currently in progress.

Japan: The reference coated particle design in Japan for the High Temperature Test Reactor (HTTR) first core is TRISO fuel particles with a low enriched (3.3-9.9%) 600 μm UO_2 kernel. The burnup design limit for the HTTR reference fuel is about 3.6% FIMA. For the HTTR second core, a current plan is to develop "advanced fuel," which will include improvements in the fabrication process and the optimization of the particle design. For example, consideration is being given to a smaller kernel and thicker buffer and SiC layers to allow for a higher burnup. The Japanese program have studied the benefits of using a ZrC layer to replace the SiC layer in TRISO coatings (e.g., Minato 2000). Studies of ZrC for DF and TF are an integral part of this fuel development program. Nuclear Fuel Industries in Japan has continued to maintain a capability to manufacture TRISO coated particles and molded fuel compacts.

Russia: The reference particle design in the Russian Federation for its version of a 200-250 MW(t) modular pebble-bed HTGR, called VGM, consists of a TRISO coated particle with a 500 μm LEU UO_2 kernel of about 8% enriched uranium. The burnup design limit for the fuel is 8.8% FIMA. There is a broad experience in reactor irradiation tests with Russian fuel, including tests with large reactivity transients. Fuel has been tested in the temperature range of 800-1600°C at up to 15% FIMA and a neutron fluence of 2.3×10^{25} n/m^2, E>29fJ (McEachern 2001). A bench-scale facility for fabrication of TRISO coated plutonium oxide particles and of fuel compacts is planned at the All-Russian Scientific Research Institute for Inorganic Materials (Bochvar Institute).

China: The Institute for Nuclear Energy Technology (INET, Tsinghua University, Beijing) has an on-going fuel development program for the design and fabrication of spherical fuel elements for use in the 10 MW(t) HTR Test Module. TRISO coated fuel particles have been manufactured with 500 μm diameter UO_2 kernels of 17% enrichment. The TRISO particles are contained in warm-pressed spherical fuel elements, using the fabrication technology developed in Germany. Preliminary operational design goals are for a maximum burnup of 9.6% FIMA, a fast neutron fluence up to $1x10^{25}$ n/m^2 and a maximum fuel temperature of 860°C. Laboratory produced particles have been irradiated to a burnup of about 1% FIMA (McEachern 2001). The INET has facilities for production-scale fabrication of TRISO coated uranium oxide for making fuel bodies.

South Africa: The South African public utility company, ESKOM, is developing a demonstration module of a 400 MW(t) Pebble Bed Modular Reactor (PBMR) for a commercial electricity generating station. The PBMR is a graphite moderated helium cooled reactor using a direct cycle gas turbine. The proposed location of the PBMR is on ESKOM property within the owner-controlled boundary of the Koeberg Nuclear Power Station situated near Capetown.

By design, provision has been made to accommodate the storage of spent fuel in the buildings for the 40-year design life of the plant and thereafter for a further period, if so required. The fuel design and process technology to be applied in the PBMR will be patterned after the German pebble fuel. An application for licensing of the PBMR has been submitted to the Republic of South Africa National Nuclear Regulator (NRR) for review and approval.

3.2.2 TRISO-Coated Particle Performance Modeling

The coated-particle fuels that will be used for deep-burn and transmutation applications must satisfy the performance requirements given in Section 4, which are specified on a core-average basis. In general, fuel performance is expected to vary significantly with spatial location in the reactor core, because of the expected, large spatial and time variations of temperature, burnup, and fast fluence, all of which strongly affect fuel performance. The following strategy is used in order to determine that the fuel satisfies its performance requirements:

A fuel design is developed that has a reasonable expectation of satisfying the performance requirements. Existing performance models are used to develop the particle design, based on conservative assumptions (e.g., all fission gas generated during irradiation is released to the buffer) and conservative service conditions (e.g., simultaneous conditions of high temperature, high burnup, and high fast fluence).

The fuel is manufactured and tested according to the plan described in this document. Both differential and integral tests are performed. The differential tests are used to develop and/or reduce uncertainties in the model parameters that are used to calculate fuel performance. The independent integral tests are used to validate the performance models.

Successful integral tests are generally a strong indication that the fuel will satisfy its performance requirements when irradiated in an actual reactor core. The capsule-irradiation tests will cover the complete range of temperature, burnup, and fluence expected in the reactor core, but the service-condition envelopes for the capsule and reactor core can still be significantly different. In other words, the average failure fraction for the test capsule could be significantly different than that calculated for the reactor core using validated models. Another potential issue is that many of the capsule tests are accelerated to a significant degree, such that the time the fuel is exposed to high temperatures is considerably shorter in capsule tests than that for fuel in an actual reactor core. For these reasons, fuel performance calculations using validated models must be performed for the entire core, in order to determine if the fuel actually satisfies the specified requirements.

Several key failure mechanisms that have been observed during irradiation of coated-particle fuels are described below.

3.2.2.1 Pressure-Vessel Failure

Pressure-vessel failure occurs when the tensile stress reached in the SiC layer exceeds its strength. During irradiation, the inner and outer pyrocarbon layers undergo irradiation-induced shrinkage. This shrinkage puts the pyrocarbon layers into tension, but it also applies compressive forces to the SiC layer, such that the SiC layer should remain in compression (and not fail) if one or both pyrocarbon layers remains intact. However, if the shrinkage is too high and/or the irradiation-induced creep is too low, the resulting stresses may be sufficiently high to fail the pyrocarbon layers, and the SiC layer could go into tension. Pressure-vessel failure is of particular concern for high-burnup fuels, because of the possibility of CO formation during irradiation and the larger inventory of fission gases. For oxide kernels, oxygen is released when the heavy-metal atom undergoes fission. Some of the oxygen is consumed by oxidation of carbide phases and fission-products, but excess oxygen can react with the carbonaceous buffer to form CO under the thermochemical conditions that exist during irradiation. Oxygen potential in the particle can be controlled by use of a kernel with a low O/Pu ratio, by use of an oxygen getter such as ZrC, or by use of a kernel with a minor plutonium carbide phase. Another potential issue that may affect particle design is the additional internal gas pressure from helium produced by the alpha-decay of short-lived higher actinides (Am, Cm).

3.2.2.2 Kernel Migration

Significant temperature gradients are present across the fuel particles when the reactor is producing power. If local temperatures are also sufficiently high, oxide and carbide fuel kernels can migrate up the thermal gradient. This phenomenon is often referred to as the “amoeba effect” and can lead to complete failure of the coating system. For carbide fuels, migration is caused by solid-state diffusion of carbon to the cooler side of the kernel. For oxide kernels, migration may be caused by carbon diffusion or gas-phase diffusion of CO or other gaseous carbon compounds. Because of the potential for CO generation in high-burnup fuels, kernel migration rates must be accurately determined to properly account for this failure mechanism in the design of the driver and transmutation fuels.

3.2.2.3 Chemical Attack of the SiC Layer

Noble metals (e.g., Ru, Rh, Pd, and Ag) are produced with relatively high yields during fission. During irradiation, the thermochemical conditions are not conducive for these elements to form stable oxides, and they can readily migrate out of the fuel kernel, regardless of its composition. Reactions of SiC with Pd have been observed during post-irradiation examinations (PIEs) for a broad spectrum of kernel compositions. Although the quantity of Pd is small compared with the mass of the SiC layer, the reaction is highly localized (the Pd is typically present as nodules at the inner surface of the SiC coating), and complete penetration of the SiC layer can occur if high temperatures are maintained for long periods of time. The cumulative fission yield of long-lived and stable Pd isotopes in Pu-239 is about an order of magnitude higher than that in U-235. For this reason, this failure mechanism may be more of a concern for the driver and transmutation fuels than it was with previous designs using low-enriched uranium kernels.

Carbide kernels are generally less retentive of fission products than oxide or oxycarbide kernels. Attack of the SiC layer by lanthanide (rare earth) fission products has been observed during PIEs of carbide fuels. This corrosion mechanism must be factored into the design if heavy-metal carbide phases are used to getter excess oxygen, in order to minimize CO formation. Estimates for UCO fuel showed that the rare earths could be retained to very high burnups for a wide range of carbide contents which totally suppressed CO formation (Nuclear Technology 1977). Preliminary analysis indicates that a similar situation should exist for plutonium containing fuels (Lindemer 2002).

In addition, chemical attack of the SiC layer by CO has been observed in UO_2 particles irradiated at temperatures above approximately 1400°C. Degradation occurred near locations where the IPyC layer was cracked. The kernels of particles with degraded SiC layers were examined with an electron microprobe, which showed the presence of silicon in the form of fission-product silicides. Thermochemical calculations supported the hypothesis that silicon is transported to the kernel in the form of SiO gas, which then reacts with fission products. Although peak irradiation temperatures should remain below 1400°C, this corrosion mechanism could manifest itself during a core heatup accident, if the particle design fails to minimize CO formation.

Assuming that the mechanisms discussed above are the dominant contributors to fuel failure, the fraction of failed particles at any location in the core and at any time may be expressed as:

$$F = 1 - (1 - P_{pv})\,(1 - P_{km})\,(1 - P_c),$$

where F = failure fraction, P_{pv} = pressure-vessel failure probability, P_{km} = kernel-migration failure probability, and P_c = SiC corrosion failure probability. These failure probabilities are correlated as empirical or semi-empirical functions of the local service conditions (e.g., temperature, burnup, fast fluence, thermal gradient, and residence time). This fuel-development plan includes test programs to provide the requisite data for developing failure-probability correlation for driver and transmutation fuel, and separate test programs for validating the use of these correlations for both fuel-particle and reactor-core design. The testing programs have been prioritized to address the most important issues for the present deep-burn, transmutation applications, including:

- Performance of pyrocarbon layers at high fast neutron fluences. As discussed previously, irradiation-induced shrinkage of the pyrocarbon layers provides compressive force to the SiC layer, which plays a very important role in preventing failure of the SiC layer. However, if the pyrocarbon layers shrink too much, they will fail because of the high tensile stresses. Irradiation-induced creep counteracts the shrinkage to some degree, which helps prevent excessive failure of the pyrocarbon layers. Both the shrinkage and the creep depend on temperature and fluence, and it is important to understand their behaviors as a function of coating properties and coating process conditions.

- SiC corrosion by palladium. The probability of failure by this mechanism has been correlated for low-enriched uranium fuels, but the dependence of the corrosion rate on palladium concentration is not well understood and has not been incorporated into the existing correlations. As discussed previously, the yield of palladium from plutonium fissions is significantly higher than that from uranium fissions. If the overall corrosion rate is controlled by palladium transport, the effect of increased palladium concentration on the corrosion rate may not be significant, but this hypothesis needs confirmation through testing.

  The time dependence of these corrosion reactions is also not well established. Some corrosion data are best correlated with a linear time dependence (implying a first-order chemical reaction is controlling); other corrosion data are best correlated with a square-root time dependence (implying solid-state diffusion is controlling). Accurate determination of the time dependence is particularly important when extrapolating the results of accelerated test data to predict in-reactor performance over longer irradiation times.

- Generation of carbon monoxide. Excessive CO generation will increase the pressure-vessel and kernel-migration failure probabilities, and CO can corrode the SiC layer at accident-condition temperatures. For a substoichiometric metal oxide kernel (MO_{2-x}) or an oxycarbide (MC_xO_{2-x}), the excess oxygen generated as a function of burnup is given by

  $$O/M_o = [2 - N_B]\,[FIMA - x/2] - (N_B)\,(x/2)$$

  where O = number of excess oxygen atoms, M_o = number of initial heavy metal atoms, N_B = number of oxygen atoms bound per for every two released, and FIMA = burnup in fissions per initial metal atom. At low burnup, some of the excess oxygen may remain trapped in the kernel. However, at high burnup the kernel becomes very porous, and it is likely that nearly all of the oxygen will escape, after which it will quickly react with carbon in the buffer to form predominately CO. From the above expression, the burnup $FIMA_{CO}$ at which CO formation begins is given by

  $$FIMA_{CO} = x/(2 - N_B).$$

- At high burnup, the kernels are highly porous and not very retentive of gases and the quantity N_B is determined largely from thermochemical considerations under these conditions. The oxygen potential in the kernel will depend on initial content of the carbide phase (the parameter x), the temperature, and the mix of fission products. Because of differences in the mix of fission products, the quantity N_B is lower for plutonium fuels than it is for uranium fuels. For example, thermochemical calculations indicate N_B is in the range 1.15 to 1.38 for Pu-239, but the range is 1.60 to 1.87 for U-235. The parameter x is currently specified at 0.5 ± 0.2, in order to prevent lanthanides from escaping the kernel and chemically attacking the SiC layer. Assuming $x = 0.5$ and $N_B = 1.15$, CO formation would begin at approximately 59% FIMA, which is well below the peak design burnup of about 80% FIMA.

  A design option that will virtually preclude CO formation is to add a thin layer of ZrC around the kernel. Excess oxygen then preferentially oxidizes the ZrC to form ZrO_2. Some limited testing of this design with UO_2 kernels out to burnups in the range 21% to 27% FIMA appeared to be quite successful. This design option is especially attractive for the transmutation fuel, since it may be difficult to incorporate Americium into a stable carbide phase (Lindemer 2002).

3.3 Fission Product Transport Technology

The status of the technology currently available to predict radionuclide transport in HTGR cores is summarized in three subsections. Subsection 3.3.1 describes the fuel performance and radionuclide release codes used for reactor core design and safety analysis. Subsection 3.3.2 summarizes the material property database from which the input correlations for these codes were derived. Subsection 3.3.3 describes the previous efforts to validate these codes by comparing code predictions with the observed radionuclide transport behavior in operating reactors and test facilities.

The scope of this technology plan is restricted to methods and data for radionuclide release from the fuel elements in the reactor core and in a geologic repository. Radionuclide transport in the power conversion system and in the containment building must also be predicted for design and safety analysis. Comparable technology has been developed for these purposes as well and will ultimately be used in the design and licensing of the DB-MHR. The current status of the general plant design technology, which is beyond the scope of this plan, has been described elsewhere (IAEA-TECDOC-978 1997, Hanson 2002).

3.3.1 Computer Codes

The computer codes currently available to predict fuel performance and fission product transport in HTGR cores are listed below, and are categorized by primary Core Performance Codes and secondary Support Codes. As the technology programs described in this plan are completed, material property correlations and component models specific to DF and TF fuel will be derived from the test data and incorporated into these computer codes for the design and safety analysis of the DB-MHR core. In addition, this program will generate independent integral test data for the validation of the DB-MHR versions of these fuel/fission product codes. The specific Design Data Needs for code improvement and validation and the attendant test programs are described in Sections 5 and 6, respectively.

3.3.1.1 Core Performance Codes

SURVEY (Pfremmer 2002): An analytical/finite-difference, core-survey code that calculates the steady state, full-core, fuel particle failure, and the full-core fission gas releases rates. An automatic interface with the core physics codes provides burnup, fast fluence and temperature distributions; likewise, the temperature and fuel failure distributions calculated by SURVEY are passed on to the metallic release code TRAFIC.

SURVEY/HYDROBURN (Pfremmer 2002): An optional subroutine in SURVEY which calculates the corrosion of fuel element graphite and the hydrolysis of failed fuel particles by coolant impurities, particularly water vapor. Transport of water vapor through the graphite web of the fuel element is modeled as a combination of diffusion and convection due to cross-block pressure gradients. The effects of catalysts and burnoff on the graphite corrosion kinetics are modeled.

TRAFIC-FD (Tzung 1992a): A core-survey code for calculating the full-core release of metallic fission products and actinides. TRAFIC is a finite-difference solution to the transient diffusion equation for multi-hole fuel element geometry with a convective boundary condition at the coolant hole surface. The effect of fluence on graphite sorptivity is modeled explicitly. The temperature and failure distributions required as input are supplied by an automatic interface with the SURVEY code.

COPAR-FD (Tzung 1992b): A stand-alone code as well as a subroutine in the TRAFIC-FD code which calculates the transient fission product release from failed and intact coated particles with burnup-dependent kernel diffusivities. COPAR-FD is a finite-difference solution to the transient diffusion equation for multi-region spherical geometry and arbitrary temperature and failure histories.

SORS (Cadwallader 1993): A core-survey code for calculating the transient releases of gaseous and metallic fission products; the code is used extensively for the analysis of core conduction cooldown transients. The transient core temperature distributions required as input are supplied by an automatic interface with a suitable, transient thermal analysis code, such as SINDA/FLUENT. SORS uses the same material property correlations that are used by the SURVEY code for normal operation but uses a fuel performance model that was specifically developed for core conduction cooldown conditions.

3.3.1.2 Support Codes

FUEL (Bennett 1992): A code that performs Monte Carlo calculations of fuel particle "pressure vessel" performance for fuel particle design and product specification development. FUEL uses a simplified, spherically symmetric, thick-walled shell stress analysis model to determine the failure probability of a statistical sample of fuel particles under constant irradiation conditions.

PISA (Pelessone 1993): A one-dimensional, spherically symmetric, coupled, thermal-stress finite element code used for fuel particle design, specification development, and capsule analysis. PISA performs deterministic, non-linear stress analysis of fuel particle "pressure vessel" performance for arbitrary irradiation histories. PISA can also be used to perform Monte Carlo calculations.

CAPPER (Bradley 1992): A code (Capsule Performance) which calculates coated particle failure and fission gas release for irradiation test capsules. CAPPER also models fuel performance for out-of-reactor tests that simulate HTGR accident conditions. It has the capability of modeling test conditions (temperature, burnup, fluence, and dimensions) that vary arbitrarily with time and position.

3.3.2 Material Property Data

The material property database provides quantitative values for attributes that provide the input to the codes listed above. For all such data, a seminal reference has been developed through an IAEA working group, and provides an encyclopedia of HTGR fuel performance and fission product transport data (IAEA-TECDOC-978 1997). The reference GA material property correlations and component models for fuel performance and fission product transport are documented and controlled in the Fuel Design Data Manual (Myers 1987).

3.3.2.1 Radionuclide Release from Fuel Kernels

There is an extensive international database on the release of fission gases from HTGR core materials. Fission gases are completely retained by intact TRISO fuel particles. For those particles that have defective or failed SiC coatings, the radiologically important short-lived gases, including I-131, are retained by the outer pyrocarbon coating, even at the peak fuel temperatures that occur during depressurized core conduction cooldown events. Consequently, the dominant sources of fission gas release during normal operation and licensing basis events (LBEs) are (1) heavy-metal contamination (tramp heavy metal) outside the particle coatings and (2) exposed fuel kernels which occur from inservice coating failure.

The present data base for fission gas release from heavy-metal contamination and from failed particles under irradiation is derived primarily from TRIGA measurements on fuel-compact matrix doped with uranium and on laser-failed fuel particles from capsules (Haire 1974, Homan 1978, Myers 1980, Myers 1984, Stansfield 1985, Myers 1987). Isothermal, inpile hydrolysis tests for the reaction of exposed kernels with water on LEU UCO fuel (HRB 17/18) were investigated at ORNL (Myers 1992). The temperature dependence of gas release from both unhydrolyzed and hydrolyzed LEU UCO fuel was addressed in the HFR B1 test which was conducted in HFR Petten in the Netherlands (IAEA-TECDOC-978 1997). These test indicate no strong burnup dependence for fission gas release from LEU UCO kernels up to a burnup of ~18% FIMA. Circumstantial evidence from the NPR-1/-2 irradiations of HEU UCO fuel to ~75% FIMA suggest a large burnup dependence (5-10x increases) at the higher burnup (Richards 1993). Limited German data for intermediate-to-high burnup UO_2 also suggest a burnup dependence for gas release (IAEA-TECDOC-978 1997).

The present data base for fission gas release from failed particles under dry core conduction cooldown conditions is derived largely from measurements on laser-failed HEU UC_2/ThO_2 particles; the iodine release data are exclusively from this source (Alberstein 1975, Myers 1979, Myers 1981). The limited available data indicate that LEU UCO particles are more retentive of fission gases than HEU UC_2 particles under core conduction cooldown conditions. There are extensive German data for the postirradiation heating of LEU UO_2 spheres that routinely included the measurement of 10.7-yr Kr-85 release (IAEA-TECDOC-978 1997). However, there are relatively few measurements of the release of radiologically important short-lived fission gases, such as 8-d I-131 (the test specimens have to be reactivated prior to heating to obtain such data).

While the existing international data base on gas release from U and Th fuels is extensive, there are relatively few measurements on LEU UCO fuel particles, and there are no direct measurements of the fission gas release characteristics of TRISO coated Pu fuels.

There is an extensive international database on the release of fission metals from oxide-based U and Th fuel kernels. Only silver, cesium and palladium (and perhaps other noble metals) are diffusely released to a significant degree from the fuel kernels at normal operating temperatures; the other fission metals, including radiologically important Sr-90, are only released by fission recoil. During core conduction cooldown events, some Sr is also diffusively released from the kernels in the hottest core region, but the more refractory fission metals are still completely retained. The present data base for fission metal diffusivities in fuel kernels is derived primarily from measurement on particles irradiated in accelerated capsules (Myers 1987).

There are German data for Cs, Sr and Ag retention in exposed oxide particles that were irradiated under near real-time conditions, as well as limited laboratory data on Cs release from ThO_2 kernels. There is a considerable amount of German data for diffusion of Cs, Sr and, to a lesser extent, Ag diffusion in exposed oxide-based fuels (IAEA-TECDOC-978 1997). The reference GA correlations for Cs, Sr and Ag diffusivities in LEU UCO kernels are characterized by very large burnup dependencies that need to be confirmed experimentally. There are no fission metal release data for PuO_x kernels or for the kernel compositions used in the reference DF and TF particles.

The release of plutonium, americium and curium from fuel kernels of various compositions [$(Th,U)O_2$, UO_2, UC_2 and UCO] under irradiation at high temperature (1100–1600°C) has been investigated (Foerthmann 1982, Mehner 1982, Silva 1983). The fractional releases from the kernels to the IPyC layer in intact particles were strongly dependent upon kernel composition. The actinides appear to be completely retained by the UO_2 kernel, but some release was observed from kernels containing as little as 3% UC_2. The apparent diffusion coefficient for Am in UC_2 was an order of magnitude higher than that for Pu at 1350°C.

The diffusivity of Pu in irradiated MOX pellets [$(U,Pu)O_{2\pm x}$] has been measured in the 1600–2500°C range and shown to be a function of temperature, burnup and kernel stoichiometry (Chilton 1978). However, the actinides in MOX fuel are typically fully saturated with oxygen and the actinides in substoichiometric DF and TF kernels may behave differently. There are no available data on the release of actinides from failed TRISO-coated PuO_x particles.

3.3.2.2 Radionuclide Transport in Coatings

There are considerable international data on the transport of fission products in the SiC and PyC coatings of TRISO fuel particles. Most of the data were obtained by the heating of irradiated particles, and an effective diffusion coefficient was deduced from the observed time history of fission product release. In principle, much of these data should be applicable to DB-MHR fuel; however, much of the early SiC data were compromised because the test particle batches contained unknown amounts of defective or failed SiC coatings. This would indicate an excessively large apparent diffusion coefficient. More recent international data suggest that volatile fission metals, like fission gases, do not significantly diffuse through intact SiC coatings (with the exception of Ag at high temperatures).

The present database for fission product transport in particle coatings resulted largely from diffusivity measurements for various fission products in laboratory tests (Myers 1987, Minato 2000). These data are supported by limited inpile data for Cs and Sr inferred from the results of irradiation experiments. These data imply that the effective diffusivities in SiC increase with increasing neutron fluence, presumably as a result of irradiation damage; this irradiation damage may anneal out during core conduction cooldown transients, but this effect has not been investigated.

There are limited data on the diffusive release of fission gases from BISO particles (Morgan 1977), but the relevance of these data to the transport of gases in the OPyC coatings of TRISO particles has to be confirmed. In any case, the available data indicate that the OPyC coating under MHR core conditions will retain the short-lived fission gases. Consequently, future tests should emphasize Ag transport in SiC coatings.

There are limited data regarding the diffusion of Pu, Am and Cm in PyC and SiC coatings on coated, uranium fuel particles (Baldwin 1978, Silva 1983, Minato 2000). The diffusion of these actinides in HTI and LTI PyC appeared to be relatively rapid at high temperature. The actinides were quantitatively retained by SiC to at least 1400°C; measurable releases were reported at 1600°C, but the condition of the SiC coatings in these experiments is unknown (Minato 2000). There are no data available for Pu release from failed PuO_x particles or transport in PyC and SiC coatings on coated Pu particles.

3.3.2.3 Radionuclide Transport in Matrix and Graphite

The international database for radionuclide transport in nuclear graphite is large, and in recognition of its effectiveness as a release barrier in HTGR cores. As described above for oxide-based fuel kernels, only cesium and silver nuclides migrate through the fuel element graphite at normal operating temperatures. The other fission metals, including radiologically important Sr-90 and actinides, are completely retained in the graphite during reactor operation. During core conduction cooldown events, some Sr is released from the graphite into the primary coolant in the hottest core regions, but the other less volatile fission metals and actinides are still completely retained. Moreover, the volatile metals released into the coolant in the hottest regions of the core largely resorb on the cooler fuel elements and reflector elements and are not released into the primary circuit. However, the existing data are scattered, and there are considerable differences in the transport properties for different grades of nuclear graphite. Consequently, the existing database must be screened to determine its applicability to the DB-MHR core.

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 (Hoinkis 1983, Myers 1984). The current correlation for Ag diffusivity in irradiated grade H-451 graphite was inferred from the measured Ag diffusivity in German A3 matrix (IAEA-TECDOC-978 1997).

The sorptivities of Cs and Sr on H-451 and H-327 graphites and over petroleum pitch matrix materials have been measured in the laboratory at partial pressures $>10^{-10}$ atm. The sorptivities of Cs and Sr 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. The sorptivity of pitch matrix is independent of fast. The sorptivities Cs, Sr and Ag on German thermosetting-resin matrix, including A3 matrix, have been measured and may apply to candidate U.S. resin matrix materials (Myers 1987). There are limited laboratory data that indicate the vapor pressure of Cs over graphite increases in the presence of coolant impurities and as a consequence of partial graphite oxidation.

Dragon Project data imply that Ag transport through graphite may be reduced dramatically by elevated system pressures at temperatures below ~1050°C.

The diffusivity of plutonium in unirradiated H-451 graphite has been measured up to 1350°C (Godsey 1986), and the desorption pressure of Pu sorbed on H-451 has been measured up to 1350°C (Fellows 1987). Based upon these measured transport properties, the release of Pu from the graphite into the primary coolant was predicted to be negligible for both normal operation and depressurized core conduction cooldown conditions. In addition, the sorption of PuC on H-451 graphite (Tallent 1985) and uranium diffusion in H-451 graphite have been measured (Tallent 1985). The transport properties of actinides in compact-matrix material and of Am and Cm in graphite have not been measured.

3.3.3 Validation of Fission Product Transport Methods

The validation of the reference design methods for predicting fission gas release during normal operation has been assessed by applying them to FSV, Peach Bottom, and several irradiation capsules. The noble gas release from FSV at end-of-life was overpredicted by about a factor of two; the cause of the overprediction is ambiguous. Fuel coating failure may have been overpredicted, the long-term effect of hydrolysis may be less severe than observed in lab tests, or a combination of both these effects may be the cause (Baxter 1994).

The noble gas release from Peach Bottom Core 2 at end-of-life was under-predicted by a factor of two or three. However, the dominant source of gas release was heavy-metal contamination, so not all the features of the gas release methodology were tested (Steward 1978). Both FSV and Peach Bottom Core 2 contained carbide fuel.

The fission gas release from irradiation capsules containing LEU UCO/ThO_2 fuel is generally predicted to within a factor of about five. However, these capsules operated dry, so the hydrolysis model was not tested. Moreover, there is inherent ambiguity in these data since the fuel failure fraction is not known with high accuracy independent of the gas release data.

The fission gas release predictions from the TRISO-P particles in three NPR capsules and the HRB-21 capsule were originally grossly under predicted with the reference design methods. The reasons were that (1) the FDDM failure models, which were based upon conventional 5-layer TRISO fuel, did not properly account for the coating failure mechanisms introduced with TRISO-P fuel; and (2) the FDDM fission gas release model did not account for the large burnup dependence observed for release from failed HEU UCO particles at burnups >30% FIMA (Richards 1993).

Considerable data on fission gas release from LEU UCO fuel are available from the COMEDIE BD-1 test (Medwid 1993). The Kr-85m R/B at end-of-life was predicted to within 2x , but the Xe-133 R/B was underpredicted by 5x, compared to the accuracy goal of 4x. In general, the dependence of R/B on isotope half-life was greater than predicted before the test (Richards 1993).

The validity of the transient gas release model used to analyze core conduction cooldown transients has not been rigorously assessed. The validity of the reference design methods for predicting fission gas release from Pu cores has not been assessed. There are no available data that are directly suitable for the validation of the design methods to be developed to predict fission gas release from DB-MHR cores containing DF and TF fuel.

The validity of the methods for predicting fission metal release from the core during normal operation have been assessed by applying the reference design methods to predict the observed metal release in Peach Bottom Core 2, FSV, irradiation capsules and in-pile loops (e.g., SSL1, SSL2, Idylle 03, four CPL2 loops, and R2 K13) (Myers 1987). Most of the available data are for the Cs isotopes with a small amount of Ag and Sr data. In general, the releases of fission metals were underpredicted by large factors, and in some cases, by more than a factor of ten. The cause of the underpredictions is ambiguous because the SiC defect fractions and the particle failure fractions are typically not well known. However, there is strong circumstantial evidence that the transport across the fuel compact/fuel element gap and the transport in the graphite web are not properly modeled.

Data on fission metal release from LEU UCO fuel are available from the COMEDIE BD-1 test (Medwid 1993). The value of the BD-1 metal release data was seriously compromised by the failure to perform the planned PIE of the fuel element and reflector element. The measured Cs-137/Cs-134 plateout inventories were underpredicted by nearly 30 times, using the FDDM UCO kernel release correlation. Cs release was predicted within 2x using the German UO_2 correlation, well within the specified accuracy goal of 10x (Richards 1993).

The validations of the methods for predicting fission metal release during core conduction cooldown transients have not been assessed systematically. The validity of the reference design methods for predicting fission metal release and actinide release from Pu cores has not been assessed. There are no available data that are directly suitable for the validation of the design methods that will be developed to predict fission metal release and actinide release from DF and TF fuel in the DB-MHR core.

3.4 Spent Fuel Disposal

The performance of DB-MHR spent fuel in a geologic repository has not been explicitly evaluated at this writing. However, such performance assessments have been made for the disposal of spent WPu fuel from the so-called Plutonium Consumption-Modular Helium Reactor (PC-MHR, Richards 1994) and for the disposal of spent LEU fuel from a commercial GT-MHR

(Richards 2002). Based upon these evaluations and on an earlier assessment of commercial HTGR spent fuel disposal options by ORNL (Lotts 1992), it was concluded that the preferred option for HTGR spent fuel is the direct disposal of unprocessed spent fuel elements in a geologic repository. In fact, unprocessed HTGR spent fuel elements are a nearly ideal waste form for permanent geologic disposal; the ceramic coated-particle fuel, encapsulated in nuclear graphite blocks, represents a far smaller proliferation risk and a far more robust radionuclide containment system than Zircaloy-clad, commercial LWR spent fuel. It is anticipated that the performance of spent DB-MHR fuel in a geologic repository will be comparable to that predicted for high-burnup, PC-MHR spent fuel although the burnups and fast fluences are higher for the DB-MHR.

The long-term radiological consequences of a repository containing DB-MHR spent fuel will be largely dependent on long-term performance of the coated particles and the corrosion/leach rates of the graphite fuel elements. Several researchers have concluded that graphite, pyrocarbon, and SiC are all excellent materials for isolation of high-level waste in a geologic repository. These previous studies are summarized below.

3.4.1 Corrosion Rates of DB-MHR Spent Fuel

As part of a DOE-sponsored program in the early 1980s, ORNL developed coated-particle waste forms, based on HTGR coated-particle fuel technology (Stinton 1982). Sol-gel technology with internal gelation was used to manufacture crystalline microspheres of simulated nuclear waste. The microspheres were coated with pyrocarbon and different combinations of pyrocarbon and SiC in a fluidized-bed coater. Leach tests were performed on the coated waste particles and more conventional glassified waste forms (e.g., borosilicate glass). Coated waste particles (including those coated with pyrocarbon only) leached at rates slower than could be detected by sensitive analytical techniques, including atomic absorption and inductively coupled plasma atomic emission. In contrast, radionuclide release from the glassified waste forms was readily detected.

Also during the early 1980s, the corrosion behavior of graphite, glassy carbons, pyrocarbon, and SiC was investigated by Pacific Northwest Laboratory (PNL), for the purpose of evaluating improved barriers for nuclear waste isolation (Gray 1980 and 1982). In order to obtain meaningful corrosion data in a reasonable time, tests were performed using deionized water at elevated temperatures (200 to 300°C) in a pressurized autoclave. Measurements of CO and CO_2 were used to quantify the corrosion rates. The test materials consisted of powdered graphite, glassy carbon powders, and supercalcine and ZrO_2 beads (~500 μm in diameter) coated with layers of pyrocarbon and of both pyrocarbon and SiC. For some of the graphite corrosion tests, a Co-60 gamma source was used to determine if the corrosion rates were enhanced by radiolysis. For the coated beads, no information about the coating properties (thickness, deposition temperatures, as-manufactured quality, etc.) was reported. The following results were obtained:

- At 250°C, the leach rate of graphite in deionized water was more than a factor of 10^5 slower than measured leach rates for waste glass.
- The corrosion rates of pyrocarbon and SiC were not significantly different, but were 5 to 10 times higher than the graphite corrosion rates.

- The corrosion rates of glassy carbon heat treated at 2600°C (GC-2600) and graphite were approximately the same. However, the corrosion rate of glassy carbon heat treated at 1000°C (GC-1000) was several times higher.
- Radiolysis had little effect on the graphite corrosion rate.

In (Gray 1982), the author acknowledges that the small difference in the pyrocarbon and SiC corrosion rates may be the result of poor-quality, porous SiC, i.e., the corrosion attributed to the SiC may actually be that of the underlying pyrocarbon. Additional data on well-characterized materials are needed to resolve this discrepancy.

The data that were reported by (Gray 1982) for two pyrolytic carbons and graphite were fitted to Arrhenius functions of temperature. These materials are described in Table 3-3, which also gives the fitted values for the pre-exponential and activation temperature. Figure 3-1 shows the corrosion rates as a function of inverse absolute temperature. The activation energies for graphite and PyC-2 are nearly the same and are about 50% lower than that for PyC-1. When extrapolated to lower temperatures, the corrosion rates for all three materials are very low, with PyC-1 showing the lowest extrapolated rate because of its higher activation energy.

The curves shown on Fig. 3-1 represent the data as reported in (Gray 1982), in terms of a mass-loss rate per unit area. The author states that the corrosion rates were calculated using the BET surface area. This approach is probably reasonable for the powdered materials, but is not appropriate for the coated beads. Both pyrocarbon and SiC have very small BET surface areas with very little surface-connected porosity. These materials are essentially impermeable at normal pressures, and the geometric surface area more accurately characterizes the exposed surface area.

Table 3-3. Corrosion-Rate Parameters for Pyrocarbon and Graphite

Material	**BET Area (cm^2/g)**	**Form**	**Corrosion Rate = A_0 Exp($-T_A/T$) (g/cm^2-d)**	
			A_0	**T_A**
PyC-1	360	Coated Particle	16	9,460
PyC-2	190	Coated Particle	0.12	6,490
Graphite	12,000	Powder	0.0045	6,240

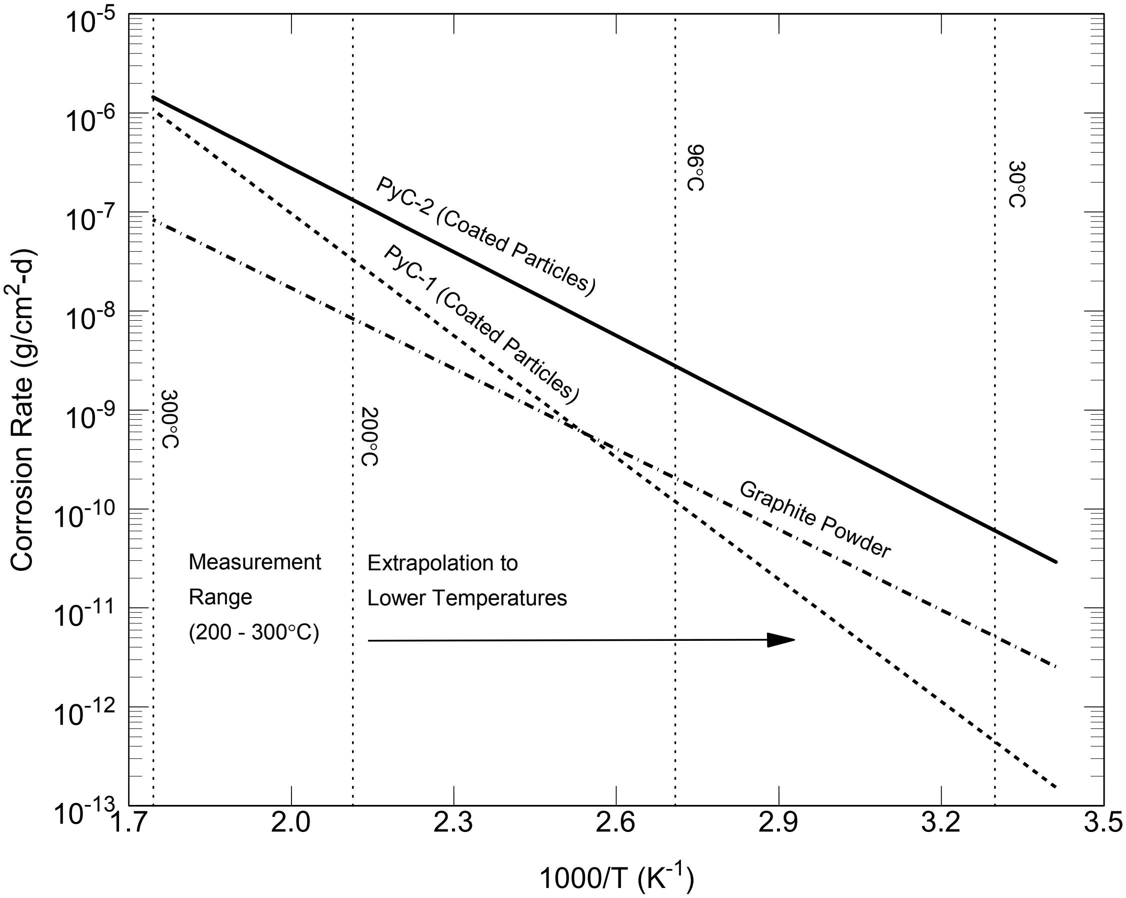

Figure 3-1. Corrosion Rates for Pyrocarbon and Graphite

For assessing performance of a coating in a repository environment, a layer-thinning rate (e.g., in the units µm/yr) is needed. The fractional mass-loss rate (FMLR) can be obtained from the (Gray 1982) data by multiplying the data by the BET surface area. The layer thinning rate (LTR) is then obtained by dividing the FMLR by the product of material density (ρ) and geometric surface area per unit mass (A_{sm}):

$$\mathrm{LTR} = \frac{\mathrm{FMLR}}{\rho A_{sm}} \quad .$$

The author (Gray 1982) did not provide the material densities or the geometrical parameters to determine A_{sm}, other than stating that the coated-particle diameters were approximately 500 µm. If it is assumed that a 500-µm particle was coated with 30 µm of pyrocarbon of density 1.9 g/cm^3, the quantity A_{sm} is 190 cm^2/g, which is about the same as the BET surface areas reported in (Gray 1982) for pyrocarbons (see Table 3-3). Hence, it is reasonable to estimate LTR by using the data as reported in (Gray 1982) and dividing by material density. Using this approach, Fig. 3-2 shows the time required to completely corrode a 40-µm thickness of the three materials described in Table 3-3.

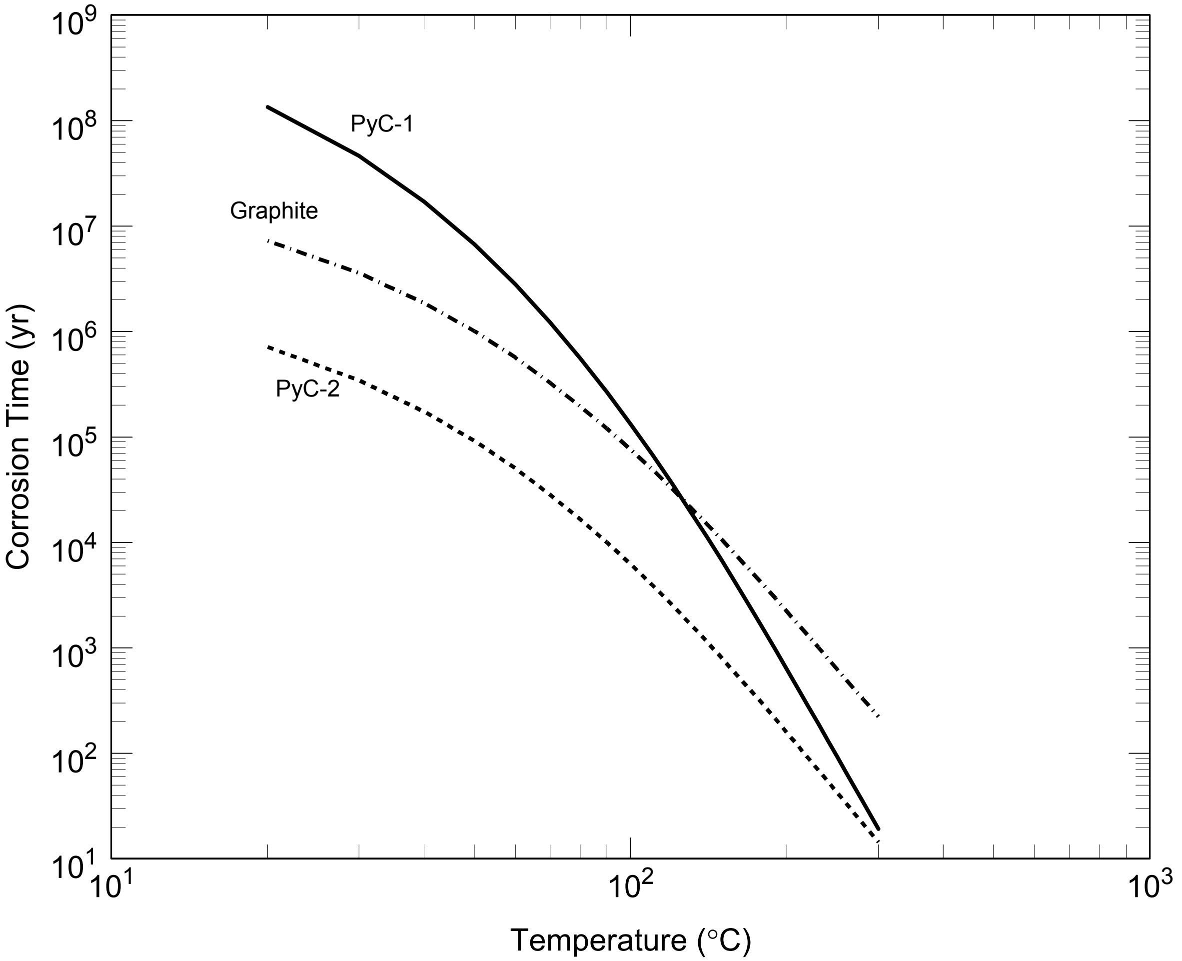

Figure 3-2. Time Required to Corrode a 40 Micron Thickness

The (Gray 1982) data also show an apparent correlation with heat-treatment temperature. Pyrocarbons, which are typically deposited at 1000 to 1300°C, exhibited corrosion rates similar to that of GC-1000. Graphite, which is typically heat treated at temperatures greater than 2000°C as part of the manufacturing process, exhibited a corrosion rate similar to that of GC-2600. This result is not unexpected, since greater outgassing occurs at higher heat treatment temperatures and the corrosion rates were inferred from CO and CO_2 measurements. Also, the corrosion rate could depend on crystallite size and structure, which can be affected by heat treatment. As part of the fuel manufacturing process, GT-MHR fuel compacts are heat treated to approximately 1800°C. Hence, the (Gray 1982) graphite corrosion rates may be representative of the corrosion behavior of both graphite and pyrocarbon in GT-MHR spent fuel elements.

The ORNL and PNL data are consistent with data obtained at the FZJ in Germany (Kirch 1990). The Germans tested the leach resistance of fuel spheres and TRISO-coated particles at temperatures up to 150ºC and pressures up to 300 atmospheres in brine for exposure times exceeding four years. Quantitative corrosion rates were not reported, but the TRISO coatings showed no indication of corrosion.

3.4.2 Long Term Performance of Graphite Fuel Blocks

Even though the graphite itself could be classified as Class C, low-level waste, it would likely represent the largest potential source term for assessing repository performance of GT-MHR spent fuel, primarily because of its relatively high C-14 content. Because the bulk of the C-14 inventory is produced from impurity nitrogen by the $^{14}N(n,p)^{14}C$ reaction, C-14 (and other activation products) can be released at a rate higher than the corrosion rate of the base material. Several tests have been performed to estimate the radionuclide leach rates from nuclear-grade graphite. Key results from these tests are summarized in the following paragraphs.

The FZJ Laboratory in Germany has examined the leach behavior of pebble fuel elements previously irradiated in the AVR (Zhang 1993). The leach tests were conducted in a synthetic brine solution for up to 464 days. Both the brine and gases vented from the top of the experimental rig were analyzed on a routine basis (approximately once per month). The following results were obtained:

- No C-14 was detected in the vented gases.
- C-14 released to the brine was chemically separated into organic and inorganic forms. Release of organic C-14 was measured throughout the test, with the release rate diminishing with time. Inorganic C-14 was released only near the beginning of the test.
- The cumulative fractional release of C-14 inventory was about 10^{-4}. The fractional C-14 release rates near the end of the test were approximately 10^{-5}/yr.

The British have performed graphite-leaching tests as part of decommissioning studies for the Magnox and Advanced Gas Reactors (White 1984)). The tests were performed on small graphite samples that were irradiated for 13 years in a CO_2-cooled Magnox reactor. The exposure conditions were not clearly defined, and it is not clear if the samples had been sealed from oxidation by the CO_2 coolant. However, prior to leaching, the samples were machined to remove the surface layer. Pre-test measurements showed significant quantities of H-3, C-14, Fe-55, Co-60, Ba-133, Cs-134, Eu-154, and Eu-155 in the graphite. Leach tests were performed using both groundwater and seawater simulants. The groundwater tests were performed at 25ºC and a pressure of 1 bar (0.987 atm) for an exposure time of 150 days. The following nuclides were detected in the leachate samples: H-3, C-14, Co-60, Ba-133, and Cs-134. Activities of other nuclides were too low to be detected. The cumulative release of all nuclides showed an initial sharp increase over the first 10 to 20 days of leaching, which was attributed to rapid release of nuclides on the graphite surface. Except for Co-60, the cumulative release of all nuclides leveled off after 40 to 60 days of exposure. The cumulative release of Co-60 continued to show a steady increase at 150 days of exposure. Results for groundwater leaching are given in Table 3-4.

Table 3-4. Radionuclide Release from Leaching Magnox Reactor Graphite

Nuclide	**Cumulative Release Fraction at 100 Days**	**Fractional Release Rate at 100 Days (yr^{-1})**
H-3	0.0048	1.5×10^{-3}
C-14	0.0014	3.3×10^{-4}
Co-60	0.013	0.039
Ba-133	0.26	0.44
Cs-134	0.18	0.044

The leach behavior of graphite irradiated in a French Magnox reactor is described in (Gray 1989). The leach tests were performed in deionized water for approximately 90 days and the results were compared with previous tests for graphite irradiated in the Hanford defense-program reactors. For Hanford graphite, the fractional release rate of C-14 was about 7×10^{-4}/yr, or about a factor of two higher than the rate measured by the British (White 1984). For the French graphite, the fractional C-14 release rates were much higher, ranging from 4×10^{-3}/yr to 0.03/yr. The higher leach rates observed for the French graphite may be caused by oxidation of the graphite by the CO_2 coolant during irradiation.

Because the DB-MHR uses high-purity helium coolant, very little oxidation occurs during irradiation, and the lower leach rates measured for British and Hanford graphite are probably more applicable. Nonetheless, if the canisters fail and the graphite is exposed continuously to groundwater, the C-14 inventory could be released over a time period that is relatively short compared with geologic time scales. For example, using the rate given in Table 3-4, the entire C-14 inventory could be released in about 3,000 years.[1]

The radiological consequences of C-14 release have been assessed for disposals of PC-MHR spent fuel (Richards Nov-1995). Release and transport calculations were performed using the REPPER code (Richards Aug-1995), using a model of the type shown in Fig. 3-3. Some of the calculations were performed using very conservative assumptions, including a flooded repository and neglecting retardation of C-14 during transport. Even with these assumptions, the predicted dose rates to an individual were very low and well below the regulatory criteria. The British also performed repository-performance assessments for their graphite and reached similar conclusions (White 1984). For example, for disposal at an inland site, the British calculated a maximum dose rate of 0.0057 mrem/yr.

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[1] It is likely that not all of the C-14 inventory would leach at a relatively fast rate. The C-14 inventory produced by neutron activation of C-13 would be part of the graphite base material and would likely leach at a much slower rate.

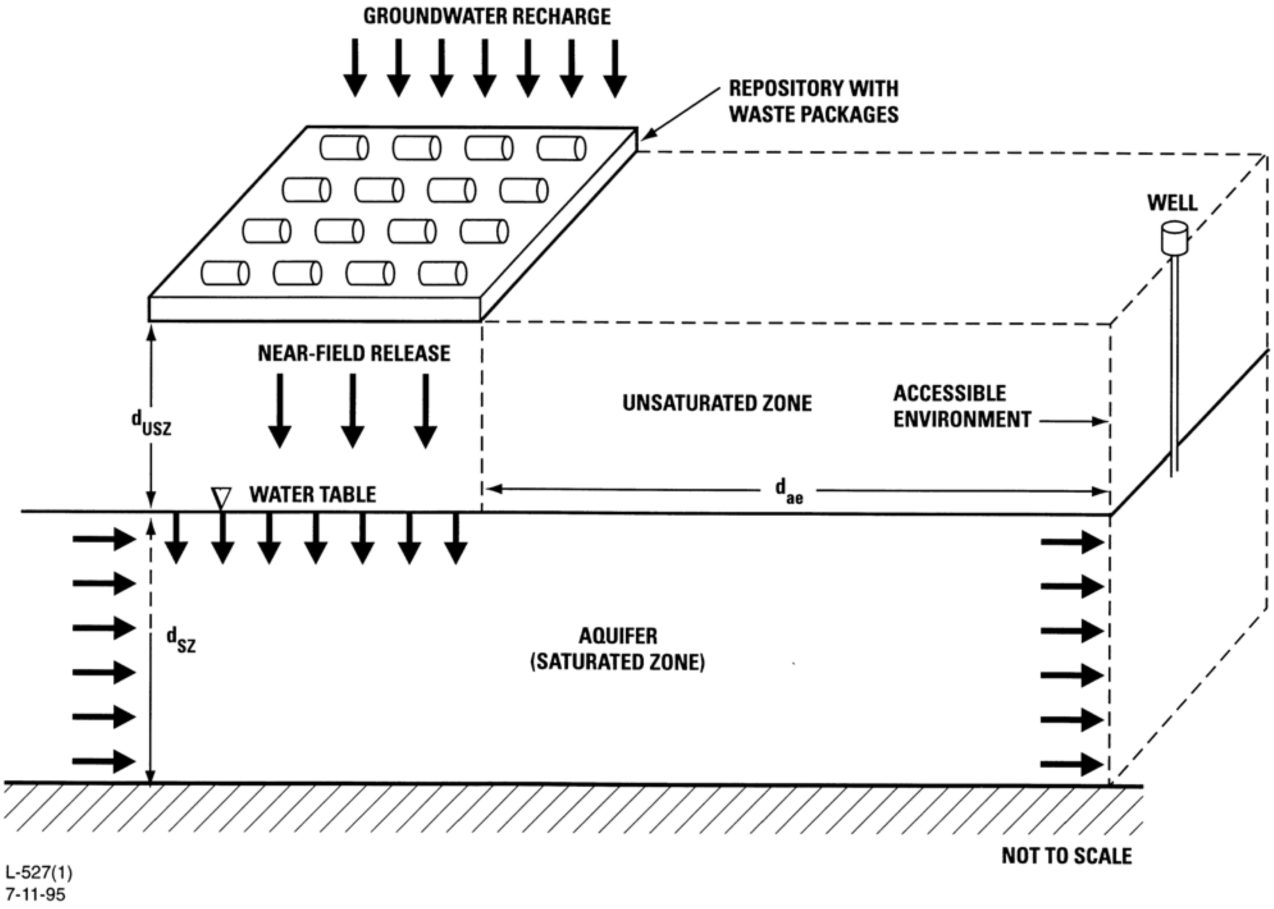

Figure 3-3. Repository Configuration and Groundwater Transport Pathways

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4. Fuel Requirements

4.1 Reactor Fuel Description

The DB-MHR is a graphite moderated, helium-cooled reactor designed for high efficiency operation and for passive safety. Highly efficient generation of electrical power is possible because of the MHR's ability to produce high temperature helium capable of driving a gas turbine that can turn an electrical generator. Passive safety is possible because of the high heat capacity provided in the core by the graphite fuel elements, and the ability of the TRISO-coated particles and the graphite core to maintain their integrity at high temperature.

The reactor core is designed to generate 600 MW(t) energy. The annular core layout is shown in Fig. 4-1. The core is made by stacking 1020 fuel assemblies in 102 columns. The fuel columns are arranged in three annular rings, with graphite moderator reflectors inside and outside the annulus and around the annulus of active fuel columns.

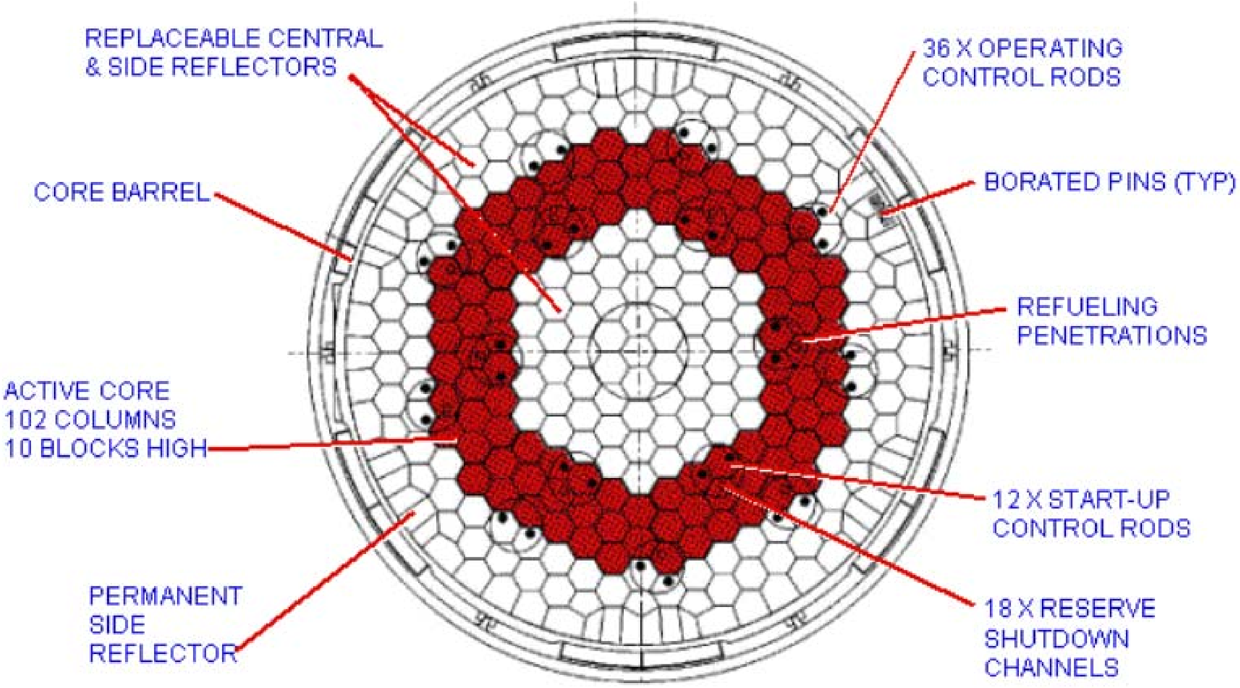

Figure 4-1. DB-MHR Core Layout

Table 4-1 presents basic parameters for the initial DB-MHR reactor core design, from (Venneri 2001), where more detail is available. Refinement of this initial design concept is being carried out in a parallel effort to the fuel development planning. Fuel management, fuel cycle length, and nuclear parameters, such as excess reactivity, reactivity coefficients, power distribution and neutron flux stability, are being evaluated. It is expected that the core design will change somewhat as a result of these analyses.

Table 4-1. DB-MHR Reactor Core Parameters

Parameter	Value
Thermal Power (MW)	600
Electrical Power from direct drive gas turbine (Brayton cycle) (MW)	285
Fuel Element Lifetime in the Core	Determine by in-service environment
Number of Fuel Columns	102
Number of Fuel Elements	1020
Reactor Arrangement	Annular - Three Rings of Fuel Columns
Power Density (kW/m^3)	~6.1
Coolant	Helium
Coolant Pressure	1025 psi (7 MPa)
Average Outlet Gas Temperature (°C)	490
Average Outlet Gas Temperature (°C)	850

4.2 Provisional Fuel Performance Requirements

Peak service conditions for the DB-MHR are defined here that are consistent with the quality and performance of coated particle fuel tested in the past, as well as the expectation of improvements from new designs and materials. These service conditions are intended to enable the DB-MHR achieve its goals of:

- Achieving a high level of fission of transuranics from LWR spent nuclear fuel in a single pass of Driver Fuel (DF) and a single pass of Transmutation Fuel (TF) through the reactor
- Using the fission energy to generate electricity efficiently with a direct drive gas turbine, and
- Utilizing one basic modular helium reactor design for the different applications of uranium-fueled commercial electrical generation, excess weapons plutonium burning, or transmutation.

These in-service bounding conditions are given as an initial guide to the fuel development and the reactor core design. These bounding conditions are needed by the fuel developers of the fuel to perform fuel design analyses, development of to prepare fuel product specifications, and to plan the details of the fuel irradiation and testing program. Core designers need this information to guide them in the trade-off inherent with optimization of the core design, such as which includes such key parameters as fuel and coolant temperatures, power densities, and fuel lifetimes. It is expected that the fuel particle designs and in-service requirements given here will be reviewed and optimized as the fuel and reactor designs progress.

Since certain coating failure mechanisms depend on the exact history of time, temperature, burnup, and fast neutron fluence, it will be necessary to define more detailed limits for combinations of these core and fuel cycle parameters as part of the overall DB-MHR design and development effort. The combinations of in-service conditions defined for DF and TF have not been fully demonstrated, even with uranium fuel. An important requirement of the fuel development workscope is to generate statistically significant irradiation data to show that the in-service failure criteria for DF and TF can be satisfied.

The limiting in-service conditions, shown in Table 4-2, indicate high burnup requirements, compatible with that demonstrated in TRISO coated plutonium oxide irradiations in the Peach Bottom-1 Reactor and the FSV HEU fuel qualification program (Miller 1985, McEachern 2001), and the service conditions from the GT-MHR design. The limits in Table 4-2 apply to the conditions in both the DB-MHR and a potential second-tier accelerator-driven sub-critical assembly. Heating of high burnup particles suggest that the temperature below which no coating failure occurs during a core conduction cooldown incident could be somewhat lower than the 1600°C demonstrated for lower burnup uranium fuels (Goodin 1985).

Table 4-2. In-Service Conditions for Driver and Transmutation Fuel

Performance Parameters	**Driver Fuel**	**Transmutation Fuel***
Core Residence Time	Determined by core design	Determined by core design
Burnup (MWd/MT)	750,000 (70% FIMA)	750,000 (70% FIMA)
Maximum Fast Neutron Fluence (E>29 fJ), (n/m^2)	8×10^{25}	8×10^{25}
Maximum Fuel Temperature (^oC):		
- normal operation	1250	1250
- accident conditions	<1600	<1600

*Limits apply to the DB-MHR and the second-tier system.

4.3 Provisional Fuel Product Specifications

The fuel management strategy for the separation of DF and TF, as described in Section 2.2, requires that the fuel particle design of the DF to be different from the design of the TF. To meet the neutronic requirements of the core, the DF kernel diameter must be relatively large and dense, while the TF kernel diameter must be smaller or it must have a lower density of TRU materials. The thicknesses of the TRISO coating layers are essentially the same for each fuel type, differing only in the thickness of the buffer layer.

4.3.1 Driver Fuel Particles

The specification for the DF particles is shown in Table 4-3. The composition of the kernel represents the greatest departure from the fuel specifications issued previously for uranium-fueled MHR designs. The specifications are proposed as a starting point for use in core design, fuel particle design, and process development tasks.

The presence of free oxygen or CO inside the TRISO coating contributes to a number of coating failure mechanisms: structural failure of the coating due to high internal pressure, kernel migration, CO attack of the SiC (Homan 1977). Therefore it is necessary to limit the oxygen content of the kernel. The kernel composition has been chosen so that the ratio of oxygen to heavy metal is less than 2. Following the experience of FTE-13, an O/Pu < 1.7 has been selected for the Driver Fuel. The behavior of the higher transuranics in coated particles is not well understood from previous experience. Alternate kernel designs for the Driver Fuel which may improve performance will be evaluated.

Table 4-3. Driver Fuel Specifications

Kernel		
Kernel Feature	Specification	
Composition	(95%) $PuO_{1.7}$/(5%) NpO_x	
Diameter	300 μm	
Density	>10	
Oxygen to Metal Ratio	<1.7	
TRISO Coating		
Coating Description	Coating Thickness	Coating Density (g/cc)
Buffer Layer	~150	<1.0
IPyC	35	1.85 - 1.9
SiC	35	3.2
OPyC	40	1.85 - 1.9

4.3.2 Transmutation Fuel Particles

The initial proposal for a specification for the TF particles is shown in Table 4-4. No kernel composition for the TF is specified, but an oxide with a reduced ratio of oxygen to heavy metal is envisioned. It is also possible to use a mixture of carbides and oxides to buffer the formation of oxygen, as has been specified for some uranium fuel applications (Potter 1972, Lindemer 2002). These alternatives will be considered during the particle design phase. In Table 4-4 the kernel specification is for a low density kernel expected to be produced in the resin-loading process which is currently the reference process.

Table 4-4. Transmutation Fuel Specifications

Kernel			
Kernel Feature		Specification	
Composition			
Element	Pu	64	Atom %
	Np	6	
	Am	28	
	Cm	2	
Diameter		250 μm	
Density		~5 g/cm^3	
Oxygen to Metal Ratio		TBD	

TRISO Coating		
Coating Description	Coating Thickness (μm)	Coating Density (g/cm^3)
Buffer Layer	>100	1.0
IPyC	35	1.85
SiC	35	3.2
OPyC	40	1.85

Initially a conventional TRISO coating is specified. However, the DF and TF may require some innovations in coating design or materials to achieve the fuel performance goals. This will be determined as the particle design proceeds. Provisions for testing more than one design for each of the particle types are included in the Fuel Development Plan.

4.3.3 Fuel Compacts

The compact consists of coated particles bonded together using a formulated thermosetting resin. Graphite powder filler is mixed with the resin in the fabrication process, and a grade of filler will be selected to control the dimensional change of the compact under irradiation. The compact will have a diameter of about 0.493 inches (12.5 mm) and a length of about 1.97 inches (50 mm). DF and TF particles will be in separate fuel compacts.

4.3.4 Fuel Element

Each fuel element is made from a machined graphite fuel block and loaded with the molded fuel compacts. The fuel block is made from nuclear-grade graphite, and is hexagonal in cross section. The dimensions are 14.172 inches (360 mm) across flats and 3.122 inches (793 mm) in length. Parallel holes, through holes for coolant and blind holes for fuel compacts, are drilled axially through the fuel blocks. Fuel blocks have three dowels to align the coolant holes in stacked blocks. Coolant holes are 0.625 inch (15.88 mm) in diameter; fuel holes are 0.5 inch (12.7 mm) in diameter. Each block has approximately 2 fuel holes per coolant hole, located on a triangular pitch of 0.740 inch (18.8 mm) from the centerline of the coolant hole to the centerline of the fuel hole.

There is a hole in the center of the block to accommodate a fuel element pickup probe for handling. Some fuel assemblies have additional holes for accommodating control rods or reserve shutdown control material. Compacts containing DF and compacts containing TF are loaded into separate holes. The number of holes with each loading type will be determined by the nuclear design.

The pattern for placing the DF and TF fuel compact stacks within the fuel block is shown schematically in Fig. 4-2. The final relative number of DR and TF stacks will be determined in the core nuclear design.

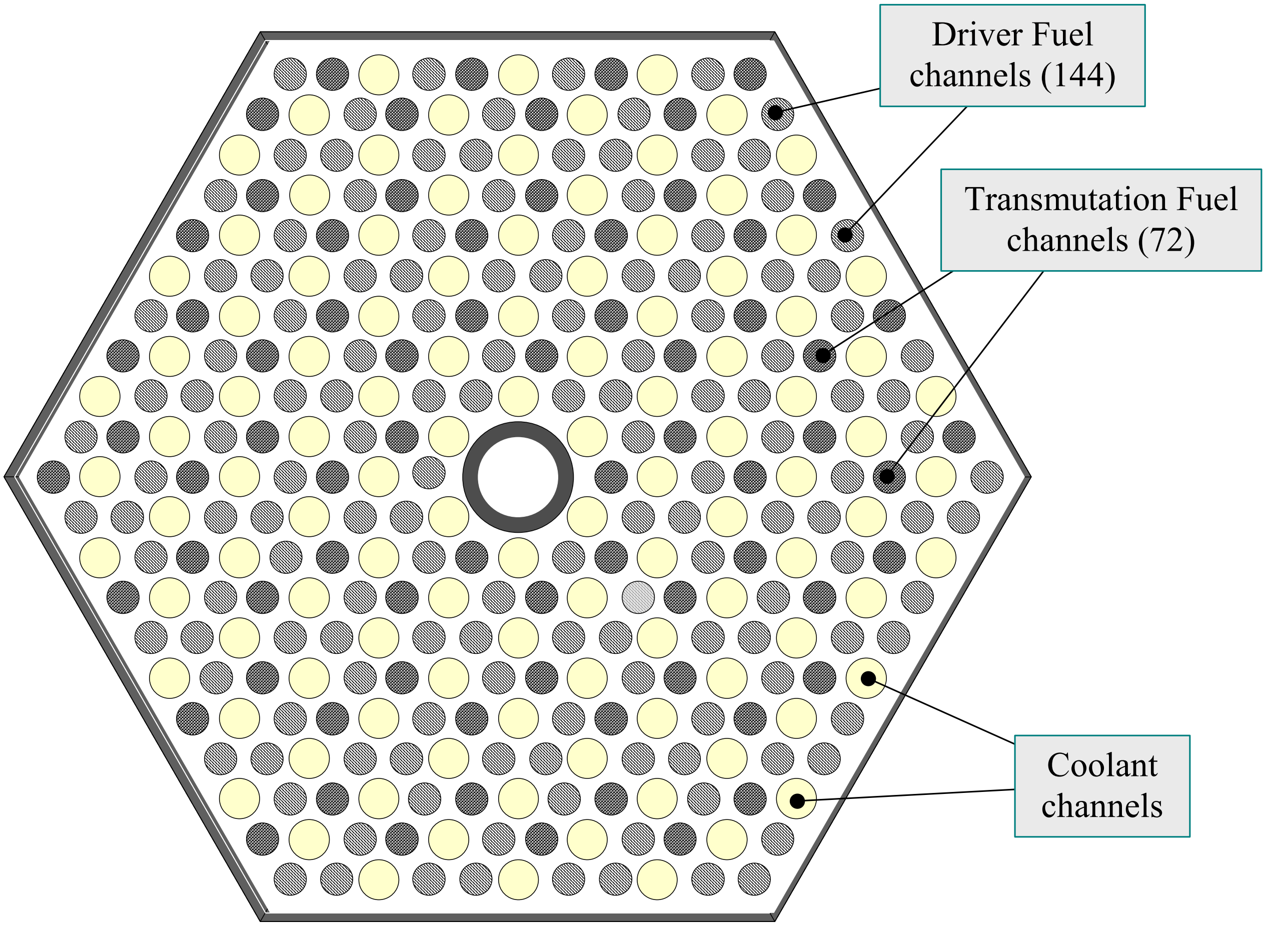

Figure 4-2. Placement of Driver Fuel and Transmutation Fuel in Fuel Element

4.3.5 As-Manufactured Fuel Quality and In-Service Performance

In order to develop a fission product source term for the DB-MHR, the retention requirements must be specified for the fuel particles and other release barriers. The sources of fission products during normal operation arise from the as-manufactured coating integrity and the in-service coating failure fractions (Goodin 1985, 1987).

Measurable particle characteristics clearly associated with the retention of fission products in the coated particles for as-manufactured fuel are the quantity of fissile and fertile material which is not contained in any coating, called “heavy-metal contamination,” and the amount of fissile material not contained in intact SiC, called “SiC defects.” These are expressed as fraction of the total amount of material in the fuel. Both nominal (50% confidence values) and limiting values accounting for uncertainties (95% confidence values) are given in the fuel product specification. Also given are limits on the quantity of coating failures that could occur during normal operation and postulated accidents.

4.3.5.1 Driver Fuel

The quality and performance requirements for the DF are shown in Table 4-5. The values are compared with those specified for the commercial GT-MHR (Scheffel 1989, Bullock 1994) The experience with FTE-13 fuel in Peach Bottom-1 Reactor suggests that the manufacturing quality achievable with that plutonium fuel can be comparable to the commercial uranium fuel requirements, so these are proposed.

The in-service failure for DF during normal operation may also be comparable to commercial fuel, but a factor of 2 higher value is proposed in the 95% confidence values to account for higher burnup and performance uncertainties.

Table 4-5. Coating Integrity Required for DB-MHR Driver Fuel

Parameter	**Commercial GT-MHR**		**DB-MHR**	
	≥50% Confidence	**≥95% Confidence**	**≥50% Confidence**	**≥95% Confidence**
As-Manufactured Fuel Quality				
Missing or defective buffer	$\leq 1.0 \times 10^{-5}$	$\leq 2.0 \times 10^{-5}$	$\leq 1.0 \times 10^{-5}$	$\leq 2.0 \times 10^{-5}$
Defective SiC	$\leq 5.0 \times 10^{-5}$	$\leq 1.0 \times 10^{-4}$	$\leq 5.0 \times 10^{-5}$	$\leq 1.0 \times 10^{-4}$
Heavy metal (HM) contamination	$\leq 1.0 \times 10^{-5}$	$\leq 2.0 \times 10^{-5}$	$\leq 1.0 \times 10^{-5}$	$\leq 2.0 \times 10^{-5}$
Total fraction HM outside intact SiC	$\leq 6.0 \times 10^{-5}$	$\leq 1.2 \times 10^{-4}$	$\leq 6.0 \times 10^{-5}$	$\leq 1.2 \times 10^{-4}$
In-Service Fuel Performance				
Normal operation	$\leq 5.0 \times 10^{-5}$	$\leq 2.0 \times 10^{-4}$	$\leq 1.0 \times 10^{-4}$	$\leq 4.0 \times 10^{-4}$
Core heatup accidents	$[\leq 1.5 \times 10^{-4}]$[a]	$[\leq 6.0 \times 10^{-4}]$	$[\leq 3.0 \times 10^{-4}]$	$[\leq 1.2 \times 10^{-3}]$

[a] Values in [square brackets] are provisional and subject to revision as the design and safety analysis evolve.

4.3.5.2 Transmutation Fuel

Transmutation Fuel will be fabricated in heavily shielded facilities where remote operations are required. Remote coated particle fabrication experience was developed in the Thorium/Uranium Recycle Facility at ORNL through the 1970s (Irvine 1965, Lotts 1977). No irradiation test data are available for TF fuels; therefore, larger safety factors are included to account for the greater uncertainty in both the as-manufactured quality and the in-service performance. The quality and performance values for TF are shown in Table 4-6.

The provisional requirements have been specified by maintaining the mean as-manufactured goals equal to the commercial GT-MHR and increasing the standard deviation in the product quality. The difference between the mean and 95% confidence level for the as-manufactured quality has been increased from a factor of 2 for the GT-MHR to a factor of 5 for the DB-MHR transmutation fuel.

The mean value for the in-service failure has been increased by a factor of 2 and the 95% confidence level has been increased from a factor of 4 for the commercial uranium fuel to a factor of 10 for TF.

The margin could be allocated between as-manufactured and in-service failure in a different manner. The method proposed here, keeping the as-manufactured quality at commercial levels, will indicate our goal of making the best fuel possible. Including the uncertainty in the in-service failure indicates that some accommodation must be made in the reactor operating conditions to account for the uncertainties associated with the transmutation fuel.

Table 4-6. Coating Integrity Required for DB-MHR Transmutation Fuel

Parameter	**Commercial GT-MHR**		**DB-MHR**	
	≥50% Confidence	**≥95% Confidence**	**≥50% Confidence**	**≥95% Confidence**
As-Manufactured Fuel Quality				
Missing or defective buffer	$\leq 1.0 \times 10^{-5}$	$\leq 2.0 \times 10^{-5}$	$\leq 1.0 \times 10^{-5}$	$\leq 5.0 \times 10^{-5}$
Defective SiC	$\leq 5.0 \times 10^{-5}$	$\leq 1.0 \times 10^{-4}$	$\leq 5.0 \times 10^{-5}$	$\leq 2.5 \times 10^{-4}$
Heavy metal (HM) contamination	$\leq 1.0 \times 10^{-5}$	$\leq 2.0 \times 10^{-5}$	$\leq 1.0 \times 10^{-5}$	$\leq 5.0 \times 10^{-5}$
Total fraction HM outside intact SiC	$\leq 6.0 \times 10^{-5}$	$\leq 1.2 \times 10^{-4}$	$\leq 6.0 \times 10^{-5}$	$\leq 3.0 \times 10^{-4}$
In-Service Fuel Performance				
Normal operation	$\leq 5.0 \times 10^{-5}$	$\leq 2.0 \times 10^{-4}$	$\leq 1.0 \times 10^{-4}$	$\leq 1.0 \times 10^{-3}$
Core heatup accidents	$[\leq 1.5 \times 10^{-4}]$	$[\leq 6.0 \times 10^{-4}]$	$[\leq 9.0 \times 10^{-4}]$	$[\leq 9.0 \times 10^{-3}]$

4.4 References for Section 4

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5. Design Data Needs

5.1 Methodology for Defining Design Data Needs

The five barriers for retaining radionuclides within the boundary of the reactor plant are: (1) the fuel kernel, (2) the TRISO coating, (3) the fuel element graphite, (4) the primary coolant pressure boundary, and (5) the reactor containment building. This initial issue of the fuel development plan for the DB-MHR addresses the first three barriers for radionuclide containment. These are the fuel kernel, the TRISO coating and the fuel element graphite. Radionuclide retention by the latter two barriers is generic to all HTGRs, although the quantitative barrier performance requirements are generally design-specific. Development programs outside the scope of the reactor core have addressed the latter two barriers (IAEA-TECDOC-988 1997).

The extent to which each of the barriers retains radionuclides during normal operation and postulated accidents must be quantified as part of the reactor design. For the past two decades, the U.S. approach to deriving radionuclide control requirements has been to use a top-down functional analysis methodology (HTGR-85-022 1985). In essence, the approach to derive the allowable radionuclide release rates from the containment building to the site boundary, and then to work “inward”, to derive in turn the allowable radionuclide releases from the primary coolant circuit, the reactor core, the coated particles and the fuel kernels. Finally, the required, as-manufactured fuel attributes are derived from the in-reactor fuel performance criteria, thus providing a logical basis for the Fuel Product Specification.

Provisional fuel performance and fuel quality requirements for the DB-MHR have been developed using engineering judgment and the results from previous functional analyses for earlier Modular HTGR designs (Hanson 2001). For this purpose, the DB-MHR was assumed to be a direct-cycle plant operating with a core outlet temperature of 850°C; the resulting requirements are summarized in Sections 4.2 and 4.3 for Driver Fuel and Transmutation Fuel, respectively. A systematic functional analysis will be done during the preliminary design phase to confirm these provisional requirements.

The reactor designer must make certain assumptions about coated-particle fuel performance and radionuclide transport behavior, especially during conceptual and preliminary design phases. In some cases, the assumption simply anticipates the expected results of a future trade study or of a more detailed analysis. In this case, the assumption is reviewed after the trade study or analysis has been completed. If the assumption is confirmed, it is replaced by the trade study, and the design is verified; if the assumption is incorrect, then the design must be modified accordingly.

In other cases, the current technology may not be sufficient to judge the correctness of the assumption at the required confidence level and this leads to a technology development need for improved technology. Conducting an R&D program typically satisfies this technology development need. Once the test program has been completed, the assumptions are reevaluated and the correctness assessed. In effect, the assumption is reduced to the first type of assumption described in the preceding paragraph. This iterative procedure is repeated until all the assumptions have been eliminated through either analysis or technology development.

On the DOE-funded MHTGR program in the mid-1980s, a formal methodology was developed for identifying technology development needs, and those have been referred to as "Design Data Needs," and that terminology is adopted here. The essence of the U.S. methodology for identifying DDNs is illustrated in Fig. 5-1.

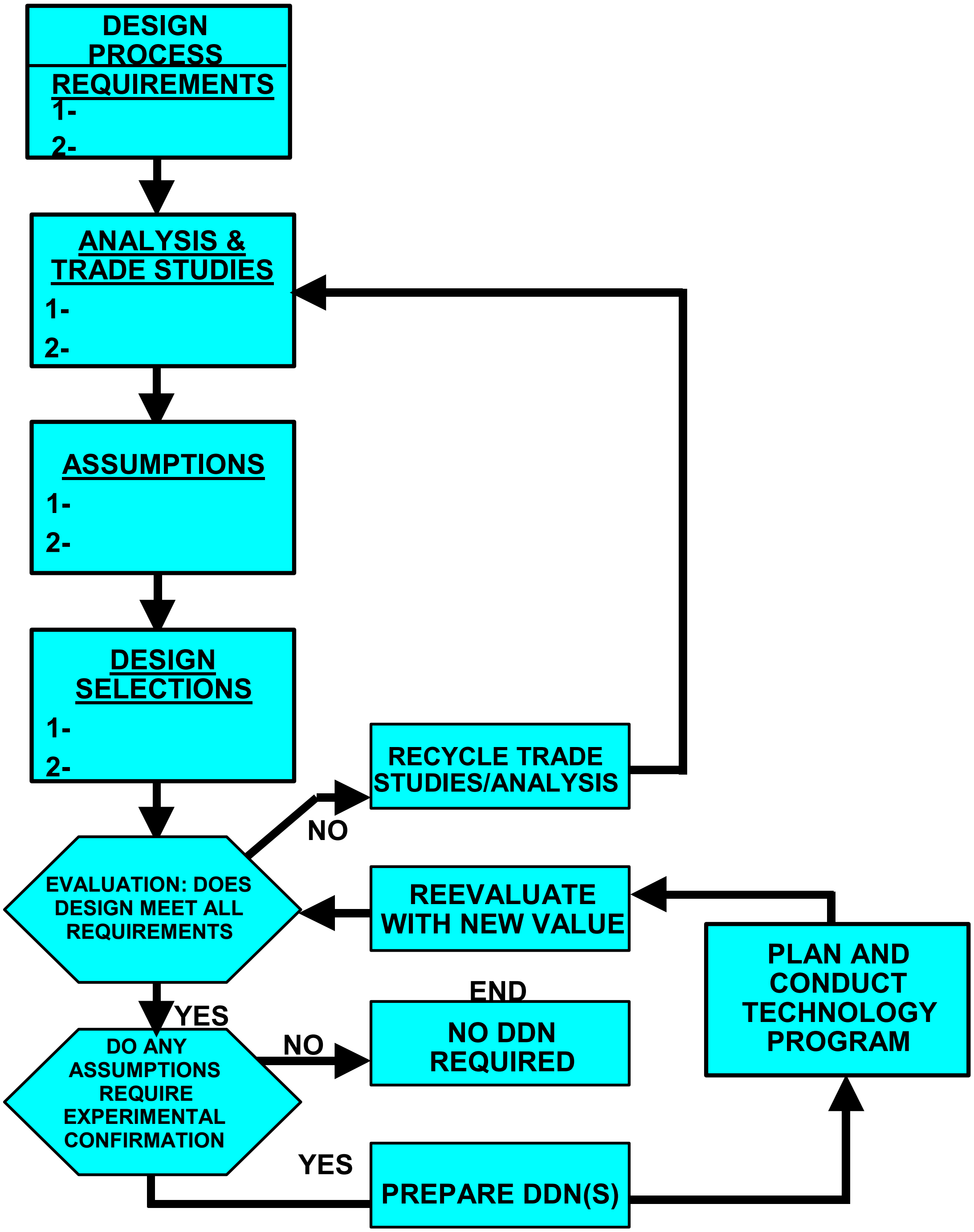

Figure 5-1. Process for Identifying DDNs

This approach proved to be a highly useful programmatic tool, especially when justifying expensive development programs to sponsoring organizations. It provided a systematic linkage between the reactor design and technology development efforts by providing logical answers to the inevitable question about why a particular test was needed to support the design or licensing. This approach has been used to identify the DDNs for the steam-cycle MHTGR, the tritium-production NP-MHTGR, the plutonium consumption PC-MHR, and the commercial GT-MHR.

A Design Data Need is a summary statement of a technology development need in a prescribed format (DDN Procedure 1986). The standard DDN format includes the following elements:

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING
    1.1 Summary of Functions/Assumptions
    1.2 Current Data Base Summary
    1.3 Data Needed
    1.4 Data Parameters/Service Conditions
2. DESIGNER'S ALTERNATIVES
3. SELECTED DESIGN APPROACH AND EXPLANATION
4. SCHEDULE REQUIREMENTS
5. PRIORITY
    Urgency:
    Cost benefit:
    Uncertainty in existing data:
    Importance of new data:

FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

REFERENCES

The contents of the various DDN sections are intuitive from the section headings, with one exception. The priority protocol requires elaboration. Each DDN is assigned a priority, which is based on the following indices:

Urgency
Cost Benefit
Uncertainty in Existing Data
Importance of New Data

The Urgency is a measure of the final scheduling date, i.e., the time in which the data are required, less the length of time needed to obtain the data. For example, if data is required in 5 years and it is estimated that it will take approximately 2 years to obtain the data, then the Urgency figure of merit would be 5 - 2 = 3. Using this approach, tasks that should be started in the current year are rated 1. The numerical rating would increase progressively, reflecting required start dates in later years.

The Cost Benefit of performing the work required to satisfy the need is a measure of the impact on a single plant cost relative to the development cost as indicated on a scale of high, medium, and low. A high Cost Benefit is defined when the savings exceed the DDN development cost by a factor of greater than ten (>10). Medium or low ratings reflect lesser Cost Benefit ratios. Note that the plant savings reflect both cost and schedule impacts.

Uncertainty in Existing Data expresses the designer's lack of confidence in the available data on which the conceptual design is based, on a scale of high, medium, and low.

Importance of New Data expresses the significance or effect of the new data on the design including consideration of available back-up solutions, on a scale of high, medium, and low.

The schedule requirements for the DDNs are stated relative to key program milestones, i.e., preliminary and final design phases. Since these key milestones have not yet been established for the DB-MHR, all Urgency indices are To Be Determined (TBD) at this writing.

5.2 DB-MHR Design Data Needs

The Design Data Needs related to fuel manufacturing, fuel performance and fission product transport for the DB-MHR are summarized below; the fission product transport DDNs are limited to transport in the reactor core (i.e., as discussed above, the ex-core, fission product transport DDNs are not included).

The source materials for developing these DDNs were those fuel/fission product DDNs and development plans prepared by GA and ORNL for the various modular HTGR designs cited above. Emphasis was placed on the DDNs for the direct-cycle GT-MHR with LEU fuel (DOE-GT-MHR-100217 1996) and the direct-cycle PC-MHR with weapons Pu fuel (Turner 1994), because they are the most directly relevant to the DB-MHR. The scope of this fuel development plan for the DB-MHR is broader than the previous fuel development plans in that this plan includes spent fuel disposal in a geologic repository as well. The point of departure for developing the spent fuel DDNs for the DB-MHR was the corresponding DDNs for the disposal of spent weapons Pu fuel elements from the PC-MHR (Hanson 1995).

The complete list of fuel/fission product DDNs for the DB-MHR is given in Table 5-1, where they are categorized by discipline: (1) fuel process development, (2) fuel materials development, (3) fission product transport, and (4) spent fuel disposal.

Table 5-1. Fuel/Fission Product DDNs for DB-MHR

DDN No.	DDN Title	Priority[(a)]
Fuel Process Development (RTW.01.XX)		
RTW.01.01	Driver Fuel (DF) Kernel Process Development	1/H/H/H
RTW.01.02	Transmutation Fuel (TF) Kernel Process Development	1/H/H/H
RTW.01.03	DF Particle Coating Process Development	1/H/M/H
RTW.01.04	TF Particle Coating Process Development	1/H/H/H
RTW.01.05	DF Compact Fabrication Process Development	1/H/M/H
RTW.01.06	TF Compact Fabrication Process Development	1/H/M/H
RTW.01.07	Quality Control Test Techniques Development	1/H/M/H
RTW.01.08	Fuel Product Recovery and Waste Treatment Technology	2/H/M/H
Fuel Materials Development (RTW.02.XX)		
RTW.02.01	Minor Actinide (MA) Loss Rates from TF Kernels during Coating	1/H/H/H
RTW.02.02	DF Coating Material Property Data	TBD/H/M/H
RTW.02.03	TF Coating Material Property Data	TBD/H/M/H
RTW.02.04	Gas Production (FP, He, CO) in DF Particles	TBD/H/H/H
RTW.02.05	Gas Production (FP, He, CO) in TF Particles	TBD/H/H/H
RTW.02.06	DF Defective Particle Performance Data	TBD/M/H/M
RTW.02.07	TF Defective Particle Performance Data	TBD/M/H/M
RTW.02.08	Thermochemical Performance Data for DF Fuel	TBD/H/H/H
RTW.02.09	Thermochemical Performance Data for TF Fuel	TBD/H/H/H
RTW.02.10	Fuel Compact Thermophysical Properties	TBD/L/H/L
RTW.02.11	Normal Operation DF Performance Validation Data	TBD/H/H/H
RTW.02.12	Normal Operation TF Performance Validation Data	TBD/H/H/H
RTW.02.13	Accident DF Performance Validation Data	TBD/H/H/H
RTW.02.14	Accident TF Performance Validation Data	TBD/H/H/H
RTW.02.15	[DF Proof Test]	TBD/H/M/H
RTW.02.16	[TF Proof Test]	TBD/H/H/H
Fission Product Transport (RTW.03.XX)		
RTW.03.01	Fission Gas Release from Core Materials	TBD/H/H/H
RTW.03.02	Fission Metal Diffusivities in Fuel Kernels	TBD/M/H/M
RTW.03.03	MA Diffusivities in Fuel Kernels	TBD/M/H/M
RTW.03.04	Fission Product Diffusivities in Particle Coatings	TBD/H/M/H
RTW.03.05	Fission Product Diffusivities/Sorptivities in Graphite	TBD/H/H/H
RTW.03.06	MA Diffusivities/Sorptivities in Graphite	TBD/H/H/H
RTW.03.07	Fission Gas Release Validation Data	TBD/H/H/H
RTW.03.08	Fission Metal Release Validation Data	TBD/H/H/H

Table 5-1. Fuel/Fission Product DDNs for DB-MHR (Cont.)

DDN No.	DDN Title	Priority[(a)]
Spent Fuel Disposal (RTW.04.XX)		
RTW.04.01	Radionuclide Leaching Rates from DF Kernels	TBD/M/H/M
RTW.04.02	Radionuclide Leaching Rates from TF Kernels	TBD/M/H/M
RTW.04.03	Long-Term Mechanical Integrity of Stressed TRISO Coatings	TBD/H/H/H
RTW.04.04	PyC Coating Oxidation Rates (Air)	TBD/L/M/L
RTW.04.05	SiC Coating Oxidation Rates (Air)	TBD/M/M/M
RTW.04.06	Resin-Matrix Oxidation Rates (Air)	TBD/L/L/L
RTW.04.07	Fuel Element Graphite Oxidation Rates (Air)	TBD/M/M/M
RTW.04.08	Graphite Noncombustibility Demonstration	TBD/H/H/H
RTW.04.09	PyC Coating Corrosion Rates (Groundwater)	TBD/M/M/M
RTW.04.10	SiC Coating Corrosion Rates (Groundwater)	TBD/H/H/H
RTW.04.11	Resin-Matrix Corrosion Rates (Groundwater)	TBD/M/M/M
RTW.04.12	Fuel Element Graphite Corrosion Rates (Groundwater)	TBD/H/H/H
RTW.04.13	C-14 Content of Matrix and Graphite	TBD/H/H/H
RTW.04.14	C-14 Leaching Rates from Matrix and Graphite	TBD/H/H/H
RTW.04.15	Chemical Impurities in H-451 Graphite	TBD/M/M/M
RTW.04.16	Radionuclide Leaching Rates from Graphite	TBD/H/H/H

[(a)] Priority protocol is Schedule Urgency/Cost Benefit/Uncertainty in Existing Data/Importance of New Data.

The schedule requirements for the DDNs are stated relative to key program milestones, i.e., preliminary and final design phases. Since these key milestones have not yet been established for the DB-MHR, all Urgency indices are To Be Determined (TBD) at this writing.

Programmatically, such classification has proven useful in the past because different organizations and, to a lesser extent, different technologies are involved in these four disciplines. A complete set of fuel/fission product DDNs for the DB-MHR in the prescribed format is included here in Appendix A.

Qualitatively, the fuel/fission product DDNs for the DB-MHR can be summarized as follows:

1. Develop and qualify the fabrication processes needed to manufacture coated particle fuel DF and TF of improved the quality and performance as defined by the fuel product specification.

2. Validate the fuel performance models that are used to predict the coated particle and fuel compact integrity for DB-MHR service conditions.

3. Reduce the uncertainty in the models and physical property data used to predict fission product transport in the core and primary coolant circuit under normal and accident conditions.

4. Validate the design methods for predicting fuel performance and fission product release from the core and transport in the primary coolant circuit during normal operation and accidents.

5. Qualify unprocessed, spent DB-MHR fuel elements for permanent disposal in a geologic repository with Yucca Mountain-type site characteristics.

The technology programs to satisfy these DDNs are presented in Section 6.

5.3 References for Section 5

DOE-GT-MHR-100217, "600 MW(t) Gas Turbine Modular Helium Reactor Design Data Needs," General Atomics, July 1996.

DDN Procedure, "DOE Projects Division Program Directive #16: HTGR PROGRAMS - Design Data Needs (DDNs) Interim Procedure," PD#16, Rev. 1, February 1986.

Hanson, D.L., et al, "PC-MHR Spent Fuel Disposal Confirmatory Test and Analysis Plan," GA Document PC-000451, Rev. 0, General Atomics, November 1995.

Hanson, D.L., "Logic for Deriving Fuel Quality Specifications," PC-000498/0, General Atomics, March 2001.

HTGR-85-022, "Procedures and Guidelines for Functional Analysis," General Atomics, June 1985.

TECDOC-988, "International Cooperation in Developing the GT-MHR," Vienna, December 1997.

Turner, R.F., et al., "Plutonium Fuel Development Plan for the PC-MHR," PC-000392, Rev. 1, General Atomics, August 1994.

6. Technology Development Programs

The formal goal of the development program is to satisfy the DDNs (Section 5) that are necessary for qualification of fuel, obtaining design data, designing an industrial-scale fuel fabrication plant and supplying fuel to an NRC-licensed DB-MHR. At the conclusion of the program: (a) a preliminary design and cost estimate will be prepared for constructing the fuel fabrication plant, and (b) the testing of prototype DF and TF will be complete. Sufficient data will be available to support a decision on detailed design and construction of a full-scale fuel fabrication plant and DB-MHR. There are additional requirements that involve the performance of the TRISO option as a component in the fuel cycle (e.g., spent fuel disposal, and compatibility with separations infrastructure).

Fuel technology development for the DB-MHR program requires close coordination of work in several specific fuel areas. The areas include (1) Fuel Fabrication Process Development [WBS 1.1], (2) Fuel Materials Development [WBS 1.2], (3) Fission Product and Actinide Transport [WBS 1.3], and (4) fuel system analysis [included as subsystems in several WBS elements]. Additionally, the fuel work must be carefully coordinated with (1) the core and reactor plant designs, (2) the separations activities, (3) the fuel cycle optimization studies, (4) the overall project licensing activities, and (5) the DB-MHR economic analyses.

Because of the overriding importance of retention of fission products and actinides in the spent fuel sent to the repository, assessment of the behavior of spent DF and TF in a repository is included as a Spent Fuel Disposal task [WBS 1.4]. The need for compatibility with a fuel separations infrastructure is not a part of this technology plan, but the likely interface requirements are outlined in Appendix B.

The top level WBS elements are summarized in Table 6-1 and the relationships between the WBS and the DDNs are shown at the beginning of each sub section. A detailed list of all WBS elements was shown in Section 2.3 (Table 2-2).

Table 6-1. Major WBS Elements for the Fuel Development Program

1. FUEL TECHNOLOGY PROGRAM			
1.1	**1.2**	**1.3**	**1.4**
Fuel Process Development	Fuel Materials Development	Fission Product Transport	Spent Fuel Disposal
1. Base Technology	1. Driver Fuel (DF)	1. Retention in Fuel	1. Long-Term Performance of Fuel Components.
2. Driver Fuel (DF)	2. Transmutation Fuel (TF)	2. Validation Tests	2. Radionuclide Transport
3. Transmutation Fuel			
4 . Prototype Analysis			

The Fuel Process Development tasks focus on the equipment and recipes (flowsheet conditions, schedules, and procedures) required to fabricate the DF and TF particles and compacts. Fuel Materials Development defines the performance of the fuel under expected service and accident conditions, to confirm that the performance requirements imposed by the reactor designer can be met. It includes measurements on test fuel performed during irradiation, post-irradiation examination, and accident condition testing.

The Fission Product and Actinide Transport task covers radionuclide retention and transport in the fuel particles, compacts, and to fuel element graphite in the reactor system. This task also includes measurements on test fuel performed during irradiation, post-irradiation examination, and accident condition testing. Given the close coupling of fuel particle failure and fission product transport to the graphite boundary (i.e., coolant channel), it is appropriate to design coordinated irradiation experiments to address both areas.

The Spent Fuel Disposal Technology examines the long term behavior of DF and TF in the repository environment both analytically and experimentally. Reaction rates of fuel components including the graphite fuel element, compact matrix, and most importantly the pyrocarbon and silicon carbide coatings on the particle. Work includes analyses of the structural integrity of the coatings, corrosion rates in air and groundwater, analyses of radionuclide migration in the repository, and leaching of radionuclides from the spent fuel in the repository.

The analyses of the fuel and fission product behavior are imbedded in each of the level 2 WBS elements. The purpose of these analyses is to confirm that the fuel system is sufficiently well understood to predict fuel behavior for normal and accident conditions.

The primary purpose of this development program is to confirm the technical basis for the fuel for the DB-MHR and to reduce the risk of the financial commitment of building a fuel fabrication facility for DF and TF and using the fuel in a demonstration reactor for transmutation of TRU.

The following sections describe the scope of work to be done, show its structure and logic, and indicate the priorities for the individual tasks.

6.1 Fuel Fabrication Process Development (WBS 1.1)

The process development tasks listed in Table 6-2 will be led by Oak Ridge National Laboratory. The DB-MHR fuels pose new challenges due to their novel actinide content (high Pu-contents, and minor actinides) and the high burnup and fast neutron fluence service limits. These characteristics will require a process development effort to support fuel testing and eventual industrial deployment. Development of these new fuels must also be supported by re-establishing a base technology that can support progress by using surrogate materials (e.g., Zr, Hf, or U) that are cheaper and easier to work with. The use of uranium in coated particles is well established and is especially important in order to re-establish a reference fuel quality.

Table 6-2. WBS elements for Fuel Fabrication Process Development

1.1 FUEL FABRICATION PROCESS DEVELOPMENT			
1.1.1	**1.1.2**	**1.1.3**	**1.1.4**
Base Technology	Driver Fuel	Transmutation Fuel	Prototype Analysis
1. Surrogate Fuel Lab	1. DF Laboratory	1. TF Facility	1. Prelim. Fab. - Plant Design.
2. Surrogate Fuel Fab.	2. DF Test Fuel Fab.	2 TF Test Fuel Fab	2. Prototype Equip. Analysis for Refab.
3. Industrial Scale Development			3. Economics of Industrial-Scale Fabrication

Beginning work with surrogate materials is part of the strategy to establish economical and less-hazardous environments before graduating to more costly glovebox (DF) and hot-cell (TF) activities. Some of the early work with TF will be conducted with small amounts of minor actinide kernels in shielded gloveboxes. This progression is also linked to the Fuel Materials Development activity, and must support the various technical screening, margin testing, and validation-testing phases that involve irradiation specimens. The final state of process development must support fabrication of validation fuel made under prototypical processing conditions. Laboratory-scale equipment will be used to produce irradiation test specimens. Most industrial-scale equipment development is conducted in parallel with the early phases of the irradiation test program.

In addition to the natural sequence of surrogate-driver-transmutation fuel development, there are additional development functions that are needed to support design of an industrial scale facility. This need is addressed in WBS 1.1.4, “Prototypic Scale Refabrication Analysis.” All of the AAA fuel fabrication laboratory processes are not readily scaled-up to industrial scale. Critical scale-up development issues will be addressed in WBS 1.1.4.1 “Preliminary Design of Industrial Scale Refabrication” and WBS 1.1.4.2, “Analyses for Prototype Refabrication Equipment.” Once reference processes have been defined for the Industrial-Scale Fuel Fabrication Plant, an analysis must be conducted to describe the scope and cost of the industrial activity. This activity is covered in WBS 1.1.4.3, “Economics of Industrial Scale Refabrication Operations,” and is one of the most important products of the development plan. It will serve as a basis for making a decision about detailed design and construction of a full-scale fabrication line. The relationship between the process development DDNs and the WBS is shown in Table 6-3.

Table 6-3. Relationship of Process Development WBS and DDNs

DDN	WBS
DF Process DDNs	
1.01 Kernel Dev.	1.1.1 Base Fuel Fabrication Technology
1.03 Coating Dev.	1.1.2 Driver Fuel
1.05 Compacting Dev.	
TF Process DDNs	
1.02 Kernel Dev.	1.1.1 Base Fuel Fabrication Technology
1.04 Coating Dev	1.1.3 Transmutation Fuel
1.06 Compacting Dev	
1.07 QC Techniques	1.1.1.1.4 QC for Uranium 1.1.2.1.4 QC for DF 1.1.3.1.4 QC for TF
1.08 Fuel Product Recovery	1.1.1 Base Fuel Fabrication Technology 1.1.4 Prototypic Scale Refabrication Analysis
1.09 Waste Minimization	1.1.1 Base Fuel Fabrication Technology 1.1.4 Prototypic Scale Refabrication Analysis

6.1.1 Base Fuel Fabrication Technology (WBS 1.1.1)

Two development functions are to be satisfied: (a) a lab-scale surrogate fuel fabrication capability to serve as a performance standard and as a development tool for the mock-up and experimentation necessary for flowsheet development, and (b) an on-going process development activity to translate the laboratory process into industrial-scale processing technology.

WBS 1.1.1.1 (Uranium Fabrication Line): This activity provides surrogate and uranium fuel compact fabrication capability. Lab-scale facilities for non-radioactive and uranium TRISO fabrication have been re-established at ORNL. Compacting and some Quality Control capabilities are being re-established to support these surrogate and uranium operations.

WBS 1.1.1.2 (Uranium Test-Fuel Fabrication): This will support flowsheet and QC optimization for the process steps that are not particularly sensitive to kernel chemistry – such as coating and compacting. TRISO coating designs for DF and TF will be tested with surrogate kernels prior to actual DF and TF irradiation tests. Adequacy of the thermochemical and mechanical design will be investigated with non-radioactive surrogates as well as depleted uranium. Enriched uranium will be used to investigate the ability of the particle design to support high burnups. HEU UCO kernels will be used to drive the particle design to high burnups, thus these fabrication capabilities will be included in the surrogate scope. The ability to make high quality (low-defect), high-burnup, high performance (low failure rate) uranium fuel is a pre-requisite for development of DF and TF test fuel.

6.1.2 Driver Fuel (DF) Fabrication Technology (WBS 1.1.2)

The majority of DF development will occur in lab-scale equipment contained in gloveboxes. The laboratory-scale compacting unit will be located first in a glovebox (for DF proof-test compacts) and then modified for use in a hot-cell (for TF proof-test compacts).

WBS 1.1.2.1 (DF Lab Facility for Fuel Fabrication) provides the DF lab-scale fabrication capability. This scope includes kernel making (from solution preparation to sintered kernel), TRISO coating, compacting, and Quality Control. Kernel making and coating will be based on previous ORNL designs for laboratory facilities (Collins 1984a, Sease, 1970). The same basic design of lab-scale coater used in the surrogate development will be adapted for use in a glovebox (as was done before at ORNL). Kernel batch sizes will range from 50-250 g Pu, and coating batch sizes will range from 25-50 g Pu. Compact formation equipment designs will be developed in coordination with General Atomics for use with ultra-pure man-made resins. Quality control activities will parallel those of the surrogates lab, except that they will be contained in gloveboxes, and will require special radiochemical analysis at certain points.

WBS 1.1.2.2 (DF Fabrication) provides for the development and production of DF test fuel. These activities will support two variations in kernel chemistry (e.g., substoichiometric Pu and carbide-containing kernels) and coating chemistry (ZrC and SiC). It is expected that modest variations in fuel design based upon feedback from fuel modeling and experiments will also be accommodated by the flexibility of the DF lab-scale facility. Pu-kernel making and Pu-coating have been done before at ORNL, so only a brief equipment shakedown with uranium is anticipated, not an extensive set of cold-tests.

In addition to the production of test fuel, the primary product of this task is the definition of optimized flowsheet conditions, operating windows, and procedures for making Driver Fuel. Among the most critical processing parameters are kernel washing, drying, sintering, and the coating conditions. Optimized flowsheet conditions for making U-Pu oxide kernels were established at ORNL in the 1980s (Collins 1984b; Tiegs 1980, Lloyd 1979), but optimized conditions for making Pu-rich kernels have yet to be defined. The limited trials done abroad (Forthmann 1977, Bairiot 1974, reviewed by McEachern 2001) and at ORNL on pure Pu-streams serve as the starting point for future process development.

Use of carbide-containing or substoichiometric kernels will require special attention to the potential for Pu volatility. The production of Pu-oxycarbide kernels is a relatively unexplored area, and extensive development efforts are needed in the oxycarbide sintering step to establish proper conditions and a uniform product. Use of additives such as ZrC to getter excess oxygen liberated when Pu is fissioned appears to be a particularly promising technique, but both process development and irradiation testing is required to demonstrate viability.

6.1.3 Transmutation Fuel Fabrication Technology (WBS 1.1.3)

Installing and operating the fully remote equipment for handling large amounts of TF is potentially one of the largest expenses and most time-consuming activities in the program. Because of this factor, the development plan incorporates a “sparse” TF fabrication concept (described below) for learning about the performance of this special fuel.

A fully dense, 200-micron diameter TF kernel contains only 50 micrograms of actinide (and only about 12 micrograms of Am and 2.5 micrograms of Cm); thus, at least a small amount of these primary fuel elements can be handled in a shielded glovebox. In this “sparse” fabrication approach, calcined kernels of design-basis TF are made in the hot-cell. The resin-loading, hot-cell equipment already exists at ORNL to make TF kernels. Small amounts of these TF kernels are removed (based upon glovebox shielding limitations) for processing in the DF glovebox equipment line. Surrogate kernels of identical sintered size are used to fill out the batch size for sintering, coating, and compacting. A fissile surrogate will be used for irradiation test fuel (either DF or HEU). The sparse TF approach permits one to get feedback on the TF design with a minimum requirement to do remote processing. The drawback is that only small amounts of TF can be handled in the glovebox – probably an insufficient quantity to permit a complete statistical evaluation of particle failure rates. However this “sparse” amount of TF is certainly sufficient for the technical screening function of phase-I irradiation testing (Section 6.2). In the period before the installation of remote coating and compacting equipment, “sparse” TF fabrication using the existing resin-loading line and the DF glovebox line will be used for the evaluation of the TF particle design.

WBS 1.1.3.1 (Establish Remote Fuel Fabrication Line). This activity provides for the installation of a remote TF fabrication capability. It will have a lab-scale coater and prototypic-scale compacting unit. The other unit operations (kernel making, solution preparation) will be of compatible scale for supplying material to these units, but are not deemed critical scale-up issues that relate to product quality. For quality control (QC) functions that require minute sampling, the QC apparatus established for the DF line will be used for QC analysis of TF. For QC that requires 100% inspection (size, shape, x-ray micrography), or larger sample sizes (burn-leach), unique hot-cell capabilities will be established.

This activity will build upon the earlier developments at ORNL directed at developing a remote fabrication capability for coated-particle fuel. During the 1960s and 1970s the interest in the Th-U fuel cycle, and thus the recycle of U-233, required remote fuel fabrication. A hot-cell complex, then called the Thorium-Uranium Recycle Facility (TURF, presently called Bldg.7930 of the Radiochemical Engineering Development Facility (REDC), was built for this express purpose (Irvine 1965, Haws 1967). Near the end of the Th-U program a particularly favorable technology for remote fabrication was developed (Lotts 1977). A resin-loading technique, dubbed the Weak-Acid Resin (WAR) process, was developed that greatly simplified the steps leading to a kernel that could be coated. A significant amount of process and equipment development was achieved (Notz 1978). These resin-derived kernels were not as dense as the sol-gel or gel-precipitation-derived kernels, but the greater simplicity of fabrication more than counterbalanced this drawback. Both the facility and basic technology will be applied to the development of a remote fabricated TF fuel. A resin-loading process has been used for over

30 years at ORNL for fabrication of kernels for use in Am/Cm-oxide CERMET heavy element production targets (Chattin 1981). Because of the new thermochemical requirements of the fuel and high burnup performance target, it is likely that a new type of resin chemistry will be necessary to fulfill the requirements of the TF kernel (Lindemer 2002).

WBS 1.1.3.2 (Transmutation Fuel Fabrication). This activity covers the fabrication of material for evaluation of TF materials. It covers both the "sparse" fabrication of TF materials in glovebox facilities for screening purposes, and the remote fabrication of materials for margin- and validation testing.

6.1.4 Prototypic-Scale Refabrication Analyses (WBS 1.1.4)

Decisions about commitment to detailed design for an industrial fabrication facility and reactor require a degree of certainty about plant design and costs. It is the purpose of this activity to answer some preliminary questions in a way that supports the handoff of the venture to the design engineering function. The products will include conceptual design flow sheets and major equipment concepts, and equipment to fabricate DF and TF, costs and schedules for the effort to design, fabricate equipment, and construct the facility.

WBS 1.1.4.1 (Preliminary Design of Industrial Scale Refabrication Plants). This activity is not intended to be purely a design function, but rather the input to the preliminary design from the development staff. The requirements for facilities to fabricate DF and TF will be defined in this activity. This activity will be carried out in conjunction with an industrial partner.

WBS 1.1.4.2 (Analyses for Prototype Refabrication Equipment). This is another cooperative activity between the development staff and the industrial partner. Not all of the final proof test material will be fabricated in prototypical equipment. The goal in this activity is to provide the development input needed to define the design path for prototypical equipment. No further laboratory development will be necessary to support the industrial design phase, but coordination with the designer is necessary. The products of this activity will include a conceptual design process flow sheet for fabrication of DF and TF, and conceptual design level designs for major process equipment. Critical design functions for scale-up and acceptance testing will be identified for the equipment.

WBS 1.1.4.3 (Economics of Industrial Scale Fabrication). This is an activity that is primarily the domain of the industrial partner. Based on the conceptual design of DF and TF fabrication facilities a cost and schedule to establish a fabrication capability will be prepared. The final report should be complete and mature enough to support a decision about going forward with large scale activities.

6.2 Fuel Materials Development (WBS 1.2)

Fuel Materials Development defines the performance of the fuel under expected service and accident conditions, to confirm that the performance requirements imposed by the reactor designer can be met. As presented in this Plan, fuel materials development includes measurements performed under irradiation, during postirradiation examination, and under conditions designed to simulate a waste repository. Except for this last element, these WBS items are organized in Fuel Materials Development, as shown in Table 6-4. This last phase, repository studies, is treated in a separate WBS.

Table 6-4. Major WBS elements for Fuel Materials Development

1.2 FUEL MATERIALS DEVELOPMENT	
1.2.1	**1.2.2**
Driver Fuel	Transmutation Fuel
1. Non-Irrad. Eval.	1. Non-Irrad. Eval
2. Irradiation Eval	2. Irradiation Eval.

The fission product (and actinide) behavior task covers retention and transport in the fuel particles, compacts, and the fuel element graphite. Given the close coupling of fuel particle failure and fission product movement to the graphite boundary, it is appropriate to design coordinated irradiation experiments to address both areas.

The fuel analysis activity ties together the fuel process development and fuel materials development tasks, and confirms that the fuel system is sufficiently well understood to justify the financial commitment required to build a reactor. The relationship between the Fuel Materials DDNs and the WBS is shown in Table 6-5.

Table 6-5. Relationship of Fuel Materials Development WBS and DDNs

DDN	DDN TITLE	WBS ELEMENT
.02.01	Minor Actinide Loss Rates from TF Kernels During Processing	1.2.2.1
.02.02	DF Coating Materials Property Data	1.2.1.1
.02.03	TF Coating Materials Property Data	1.2.2.1
.02.04	Gas Production in DF Particles	1.2.1.1
.02.05	Gas Production in TF Particles	1.2.2.1
.02.06	DF Defective Particle Performance Data	1.2.1.2
.02.07	TF Defective Particle Performance Data	1.2.2.2
.02.08	Thermochemical Performance Data for DF Fuel	1.2.1.1
.02.09	Thermochemical Performance Data for TF Fuel	1.2.2.1
.02.10	Fuel Compact Thermophysical Properties	1.2.1.1 and 1.2.2.1
.02.11	Normal Operation DF Performance Validation Data	1.2.1.2.3
.02.12	Normal Operation TF Performance Validation Data	1.2.2.2.3
.02.13	Accident DF Performance Validation Data	1.2.1.2.3
.02.14	Accident TF Performance Validation Data	1.2.2.2.3

6.2.1 Driver Fuel (WBS 1.2.1)

6.2.1.1 Evaluation of Candidate Driver Fuel Design Selections

WBS 1.2.1.1.1 (Preliminary DF Thermochemical Analysis): Extensive data exist related to the thermochemical failure modes for coated particle fuels with kernels composed of oxides and carbides of uranium, thorium and plutonium (Lindemer 1976, Miller 1985, Gruebmeier 1977, Goodin/Nabielek 1989). Thermochemical data exist for TRISO coated uranium and plutonium based fuel kernels, but no comparable database exists for DF and TF kernel compositions. Fuel performance at the very high burnup anticipated for DF and TF has not been explored in a conclusive fashion for any kernel composition with the possible exception of UC_2 to 70% FIMA for which relatively good performance under irradiation and postirradiation heating has been demonstrated (SAR 1978). Variations in both kernel chemistry and coating materials from the presently assumed baseline of substoichiometric kernel and a SiC coating may be necessary in order to meet the performance objectives. Oxycarbide kernels or ZrC-gettered kernels (Bullock 1983; Bullock 1984) will be considered as alternative designs to a substoichiometric kernel to provide for oxygen management during fission. ZrC coatings may be necessary to provide for adequate resistance to Pd-attack at high burnup and temperature. These design options will be evaluated based on their thermochemical, structural, and neutronic viability.

A number of out-of-reactor tests have been identified which can produce significant benefits to the program. Data are needed from single-effects tests to quantify the important thermochemical phenomena for TRISO-coated DF under DB-MHR service conditions for normal operation and postulated accidents. These thermochemical studies will include (a) basic studies to confirm oxygen management strategies (b) assessment of Pu volatility and potential transport, (c) studies to define Pd-attack of SiC and ZrC, and (d) tests to confirm the materials properties at high neutron exposures. Much of this work will be done with surrogate materials, and some will be analytical. These data will be used to refine the existing thermochemical performance models for use in core design and safety analysis.

WBS 1.2.1.1.3 (Compact Thermophysical Properties): Fuel compact thermophysical data have been obtained from measurements of thermal expansion, thermal conductivity, and heat capacity for HTGR fuel compacts at General Atomics as part of commercial MHR programs (Myers 1987). Other tests have been performed on irradiated fuel compacts to determine thermal conductivity versus macroporosity. However, this database is small and does not fully cover the new compacting materials and service conditions that are anticipated.

Data are needed to characterize thermophysical properties for a statistically significant number of DF and TF compacts including heat capacity, thermal expansion and thermal conductivity. Data are required as a function of shim content, fast neutron fluence and temperature to validate the core thermal design methods at a 95% statistical confidence level. Thermal conductivity characterization is particularly important and is needed for both unirradiated and irradiated fuel compacts containing TRISO-coated DF and TF particles. Simulated fuel compacts containing inert TRISO-coated particles rather than fuel particles will be used in tests to characterize the change in thermophysical properties of DF/TF compacts with irradiation. These irradiations will be conducted at the typical service conditions (temperature, fluence) anticipated for actual fueled compacts. It is planned to include some unfueled compacts (or representative portions) in the screening and margin tests to provide this test material. Capsule design will accommodate this inclusion. This approximation of compact thermophysical properties is deemed adequate for all but the final phase of capsule irradiation testing associated with final NPP plant design. Out-of-pile measurement of thermal conductivity and heat capacity of the unfueled compacts is fully supported by the in-house capability of the Low Activation Materials Development and Analysis (LAMDA) laboratory at ORNL. Linear expansion measurements will be derived from the actual compacts after irradiation. Linear expansion at temperature can be done in the LAMDA laboratory on unfueled compacts.

6.2.1.2 Driver Fuel Irradiation Tests

An irradiation test program will be conducted to provide a basis for DF design and process improvements and finally to demonstrate that performance goals for the fuel can be met. The irradiation program will have three phases: (a) screening of candidates, (b) margin testing of reference fuel designs, and (c) validation testing of reference fuel design. The irradiation program will be coordinated with the process and materials development effort to evaluate candidate TRISO designs that allow for different coating and kernel materials (based upon thermochemical, structural, and nuclear analysis). These screening tests will lead to the selection of reference fuel design. Margin tests are conducted to define the sensitivity of the reference fuel design to variations in exposure conditions and to define the ultimate fuel performance limits. The final test is a validation test that is conducted with optimized design and fabrication conditions, and with a more mature definition of prototypical irradiation conditions.

Highly resolved fuel performance statistics and performance confirmation are to be achieved at the validation stage (capability for detecting a single particle failure). Fuel performance statistical requirements are to be defined as determined for the MHR-1 Irradiation Test Specification, and by the methods defined in the NPR Fuel Development Plan (McCardell 1992). A less exacting statistical fuel performance standard will be adopted for screening and margin tests, since these tests are used to screen candidates and identify boundaries, and do not require as high a resolution. We expect the new multi-cell capsule currently being designed at the Idaho National Engineering and Environmental Laboratory (INEEL) for use in the Advanced Test Reactor (ATR) to be able to support detection of a single particle in the irradiation tests proposed in the this Plan.

Six separate irradiations are scheduled to cover the phases of screening, margin, and validation tests. The timing of these irradiations is shown in Table 6-6. Except for the earliest HEU benchmark trials, all of these irradiations will be conducted in multi-cell capsules.

Table 6-6. DF Irradiation, PIE, and Accident Condition Testing Schedule

CAPSULE	**PURPOSE**	**2002**	**2003**	**2004**	**2005**	**2006**	**2007**	**2008**	**2009**	**2010**	**2011**	**2012**	**2013**	**2014**	**2015**
HEU-1	Benchmark		xxx	xxxx	xxxx										
HEU-2	Benchmark		x	xxxx	xxxx	xx									
DF1	Screening				xxx	xxxx	xx								
DF2	Screening				xx	xxxx	xxxx	x							
DF3	Margin					x	xxxx	xxxx	xxxx	xx					
DF4	Validation								x	xxxx	xxxx	xxxx	xx		

The HEU trials will be conducted with an existing single-cell capsule design for HFIR. A design containing 4 cells per capsule is being developed and is the basis for our multi-cell irradiations.[2] ATR is the most likely irradiation facility for the multi-cell capsules, but there are also foreign reactors that could be used. For all irradiations, the capsule preparation and loading takes about one year of the time-line, and the post-irradiation examination, or PIE, (including accident tests) also take about one year. At the very minimum, the in-service performance and initial PIE results will be available before the next phase of irradiation testing is commenced. Irradiation periods vary due to the design of the experiment and the desired irradiation acceleration factor (burnup rate higher than that experienced in the power reactor). Early screening activities are conducted with a higher acceleration factor.

6.2.1.2.1 Phase I Screening of DF Candidates (FY 2003 – FY 2008)

WBS 1.2.2.2.1.1 – Base Technology**:** It is important to obtain an early demonstration that the TRISO particle (kernel, coatings) and compact process technology has been re-established at ORNL to serve as a fuel quality benchmark. In order to accomplish that task simulated DF in the form of TRISO coated HEU will be irradiation tested to peak reactor design burnup and temperature. This work will first be carried out in a DF capsule **(HEU 1)** that will contain only unbonded simulated DF particles. The use of unbonded particles will permit a more rapid Post Irradiation Examination (PIE) of particle condition after irradiation than would be possible with compacts and also avoids the delay associated with getting compact fabrication equipment into operation. In addition, the unbonded particles will provide samples that can be easily utilized in the initial Accident Condition Tests. This fuel will be irradiated to 75% peak burnup at 1250°C and $8\text{x}10^{25}$ n/m^2 ($E \geq 29$fJ). This first capsule will use proven HFIR capsule design technology and will not require development of a new capsule design. Spikes in the capsule purge gas

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[2] The AGR program is planning to utilize a multi-cell capsule design with up to six independently purged and operated cells. If this design proves viable, it will be considered for use on this program as well. The tradeoff with regard to the number of cells is greater flexibility and diversity in test specimens and test conditions versus smaller sample sizes (i.e., numbers of particles) and greater mechanical complexity (with attendant reliability issues).

activity during irradiation will signal any particle failure. Successful demonstration of good performance in this fuel will allow the more difficult process development effort on Pu bearing fuel to proceed with added confidence that high quality TRISO coatings meeting performance criteria are being applied.

A second capsule, designated **HEU-2** will also use simulated DF particles containing HEU in compacts made in the laboratory scale compact equipment. It will be also be fully instrumented, and irradiated in HFIR under the same conditions of HEU-1. Separate measurement of bonded and unbonded particles will also indicate whether the compacting operation is a significant source of fuel failure.

WBS 1.2.1.2.1.2 (Candidate Screening): In two multi-cell capsules there will be candidate DF designs containing different fuel materials: SiC vs. ZrC coating, substoichiometric vs. alternative kernel chemistry (oxycarbide or ZrC-getter design). The purpose of these tests will be to compare performance of the candidates and select the TRISO coating with the best performance. The **DF1** will contain unbonded TRISO particles, and **DF2** fueled compacts. After completion of PIE and Accident Condition Tests on the irradiated particles the data will be analyzed and a selection of reference design will be made between the candidate designs. Follow-on capsules will only irradiate the reference DF design. Design options will be based on the results of ongoing thermochemical, structural, and nuclear analysis.

6.2.1.2.2 Phase II Margin Testing of Reference DF (FY 2006 - 2010)

WBS 1.2.1.2.2.1 (Margin Test): Irradiation test **DF3** is a multi-cell capsule margin test that will focus on demonstration of the performance of DF compacts under the most severe DB-MHR core temperature, burnup, and fast neutron exposure. The range of exposure conditions will be defined to permit evaluation of the limits which correspond to severe fuel failure due to the primary structural (i.e., due to fluence effects) and thermochemical failure mechanisms. Additional information about the short duration failure modes at extreme temperatures will be attained at the end of irradiation period. Two chambers in the multi-cell capsule will be devoted due to evaluating the structural failure mechanism at high fluence – one chamber is devoted to a low temperature condition (800°C), and the other will be held at the nominal operating temperature (~1050°C) that will avoid thermochemical failure modes. The other two cells are configured to track the design basis fuel temperature evolution. One cell will be at the design basis temperature, and the other will be significantly beyond the design basis temperature.

A provisional margin test matrix is summarized in Table 6-7. The service conditions for this margin test will be reevaluated and better quantified when the test specification is prepared in consideration of the then available results from the DF1 and DF2 irradiations and the core analyses that will have been performed. Traditionally, a margin test would include service conditions (e.g., fast fluence and/or temperature) that are sufficiently severe to cause 0.001 to 0.01 failure fractions (i.e., sufficiently high that the failure mechanism(s) can be reliably detected by metallography, etc.); bounding core service conditions should not produce that level of failure. Stated differently, a properly designed margin test should find the performance "cliffs."

Table 6-7. Provisional Margin-Test Matrix for DF and TF

	STRUCTURAL FAILURE	THERMOCHEMICAL FAILURE
Low Temperature	~ 800°C (steady)	Design Basis Temperature (tracking)
High Temperature	~ 1050°C (steady)	Beyond Design Basis Temperature (tracking)

6.2.1.2.3 Phase III Fuel Validation (FY 2009 – 2013)

WBS 1.2.2.2.3.1 (Validation Test): Reference design DF compacts, fabricated with an optimized design and using optimized processes, will be available at this stage of the program. A validation irradiation test, **DF4,** will be conducted to demonstrate that under normal operating conditions such fuel performs as predicted by the fuel performance models developed using previous data. The DF4 capsule will expose fuel to conditions simulating DB-MHR core average temperature (800°C), and core peak temperature (1250°C). The design burnup will be 80% FIMA and $9x10^{25}$ n/m^2which is about 10% beyond expected peak DB-MHR burnup and fast neutron exposure. Highly resolved fuel performance statistics and performance will be achieved in this test by pairing the symmetrical cells in the multi-cell capsule (with the capability of detecting a single particle failure).

6.2.2 Transmutation Fuel (WBS 1.2.2)

6.2.2.1 Evaluation of Candidate Transmutation Fuel Design Selections

Many of the elements of this activity are analogous with the DF development work described in WBS 1.2.1. Similarities are noted, and distinctions are emphasized.

WBS 1.2.2.1.1 (TF Thermochemical Performance): Kernel Composition**:** The TF kernel has a heavy metal content of 65 atom% Pu, 5 atom% Np, 25 atom % Am, and 5 atom % Cm. The radioactivity of gram quantities of TF will require remote processing at all stages and is thus much more costly in terms of facilities and manpower.

Kernel Fabrication**:** The same thermochemical studies will be done for TF as were described earlier for DF. In addition the issue of Am/Cm volatility will also be addressed.

TRISO Particle Failure: The same consideration for the driver fuel discussed earlier applies here, but the contribution of helium generation during transmutation must be added.

Fission Product Transport: The same considerations discussed for the DF applies here, but with the addition of the elements produced from transmutation of the Cm and Am.

WBS 1.2.2.1.2 (TF Fuel Compact Thermophysical Properties): The issues discussed earlier for the DF compacts apply here.

6.2.2.2 Transmutation Fuel Irradiation Tests

An irradiation test program will be carried out to provide a basis for TF design and process improvements and finally to demonstrate that performance goals for the fuel can be met. The irradiation program will be coordinated with the process and materials development effort to evaluate SiC and ZrC as the DF TRISO barrier coating and select the reference TRISO design. Following selection of the reference TF design, irradiation tests will be conducted to qualify TF TRISO fuel for DB-MHR service prior to production line scale-up. Since the TF is starting from a much smaller experience base than DF, and it is more difficult to fabricate, the irradiation of TF and post irradiation examination are planned to finish after that of DF.

The irradiation experiments will expose fuel to a range of conditions including those more severe than expected DB-MHR core average conditions. Enough particles will be irradiated in each capsule to provide valid statistical evidence that the DB-MHR core average fuel failure criteria can been met. In a manner analogous to the DF irradiations four separate irradiations are scheduled to cover the phases of screening, margin, and validation. The timing of these irradiations is shown in Table 6-8. All of these irradiations will be conducted in multi-cell capsules. A design containing 4 cells per capsule is being developed and is the basis for our multi-cell irradiations.

Table 6-8. TF Irradiation, PIE, and Accident Condition Testing Schedule

CAPSULE	PURPOSE	2002	2003	2004	2005	2006	2007	2008	2009	2010	2011	2012	2013	2014	2015
TF1	Screening					xxxx	xxxx	xxxx							
TF2	Screening					xx	xxxx	xxxx	xx						
TF3	Margin							xx	xxxx	xxxx	xxxx				
TF4	Validation										xx	xxxx	xxxx	xxxx	

6.2.2.2.1 Phase I: Screening of TF Candidates (FY2006-2009)

In two multi-cell capsules there will be candidate TF designs containing different fuel materials, substoichiometric vs. alternative kernel chemistry (oxycarbide or ZrC-getter design), and resin-loaded designs with various diluents. The purpose of these tests will be to compare performance of the candidates and select the TRISO design with the best performance. The **TF1** will contain unbonded TRISO particles, and **TF2** will contain fueled compacts. After completion of PIE and Accident Condition Tests on the irradiated particles the data will be analyzed and a selection of reference design will be made between the candidate designs. Follow on capsules will only irradiate the reference DF design. Design options will be based on the results of ongoing thermochemical, structural, and nuclear analysis.

6.2.2.2.2 Phase II: Margin Testing of Reference TF (FY 2008 to FY 2011*)*

WBS 1.2.2.2.2.1 (Margin Tests): **TF3** is a multi-cell capsule margin test that will focus on demonstration of the performance of TF compacts under the most severe DB-MHR core temperature, burnup, and fast neutron exposure. The range of exposure conditions will be defined to permit evaluation of the limits which correspond to severe fuel failure due to the primary structural (i.e., due to fluence effects) and thermochemical failure mechanisms. Additional information about the short duration failure modes at extreme temperatures will be attained at the end of irradiation period. Two chambers in the multi-cell capsule will be devoted due to evaluating the structural failure mechanism at high fluences – one chamber is devoted to a low temperature condition (800°C), and the other will be held at the nominal operating temperature (~1050°C) that will avoid thermochemical failure modes. The other two cells are configured to track the design basis fuel temperature evolution. One cell will be at the design basis temperature, and the other will be significantly beyond the design basis temperature, but below known failure limits for the fuel materials.

A provisional margin test matrix is summarized in Table 6-7. As with the DF margin test, the service conditions for the TF margin test will be reevaluated and better quantified when the test specification is prepared in consideration of the then available results from the TF1 and TF2 irradiations and the core analyses that will have been performed. Traditionally, a margin test would include service conditions (e.g., fast fluence and/or temperature) that are sufficiently severe to cause 0.001 to 0.01 failure fractions (i.e., sufficiently high that the failure mechanism(s) can be reliably detected by metallography, etc.); bounding core service conditions should not produce that level of failure. Once again, a properly designed margin test should find the performance "cliffs."

6.2.2.2.3 Phase III: Fuel Validation Test (FY 2011 to FY 2014*)*

Reference design TF compacts, fabricated with an optimized design and using optimized processes, will be available at this stage of the program. A validation irradiation test, **TF4,** will be conducted to demonstrate that under normal operating conditions such fuel performs as predicted by the fuel performance models developed using previous data. The TF4 capsule will expose fuel to conditions simulating DB-MHR core average temperature (800°C), and core peak temperature (1250°C). The design burnup will be 80% FIMA and $9x10^{25}$ n/m^2 which is about 10% beyond expected peak DB-MHR burnup and fast neutron exposure. Performance data that are statistically significant will be obtained in this test by pairing the symmetrical cells in the multi-cell capsule (capability for at least 10^{-5} failure fraction resolution).

6.3 Fission Product and Actinide Transport (WBS 1.3)

The fission product and actinide transport work scope in support of DB-MHR design and licensing is planned to be consistent with the overall program goal of providing a proof-of-concept by the year 2010. DDNs RTW 03.01 - 03.08 provide definition of the required data. The primary objective is to determine if DF and TF fuels present any significantly new radionuclide release behavior beyond that already observed with uranium fuels.

The work scope addressed herein is limited to radionuclide transport in DF/TF kernels, particle coatings, fuel-compact matrix and fuel-element graphite. Radionuclide transport in the primary coolant circuit and in the reactor containment building is not addressed in this FDP. Nevertheless, credit for radionuclide retention by primary coolant pressure boundary and the reactor containment building will necessarily be taken in the design and licensing of the DB-MHR in order to meet radionuclide control requirements at the specified confidence levels. It is assumed that other on-going U.S. and international HTGR technology programs will address in a timely fashion the DDNs associated with these ex-core radionuclide release barriers.

Initially, a chemical evaluation of DF and TF kernels as a function of burnup and temperature will provide an indication of the expected fission product forms for a range of kernel compositions. When the kernel selection is sufficiently defined for irradiation testing in the Fuel Materials work scope (Section 6.2), the radionuclide transport work outlined here will also be initiated so that the kernel retention (fission product and actinide) can be investigated in parallel with general coating integrity tests. High kernel release, even with good coating performance, can result in unacceptable accident releases from damaged particles and corrosion of intact SiC coatings by fission products, such as palladium. Even though kernel retention is reduced at high burnup for gaseous and volatile fission products, retention of the actinides and fission products that form refractory compounds should be substantial and important.

A considerable amount of work has been done on coated-particle uranium fuel (Section 3.3), and this program will take full advantage of these existing data. However, significant differences in the kernel chemistry for DF and TF will likely influence the transport of fission products and actinides. These include different kernel oxygen potentials, somewhat different yields of fission products, possible volatile actinide kernel constituents, and a much higher yield of fission products known to be corrosive to SiC (e.g., Pd isotopes). Moreover, the target peak burnup and fast fluence for the DB-MHR are well beyond the current database for fission product transport.

In-pile irradiations will be performed on compacts containing DF and TF particles, with on-line gas monitoring and post irradiation examinations providing a quantification of fission product and actinide retention. The radionuclide retention by the compact matrix and moderator graphite will be determined by irradiation testing of simulated fuel bodies, which are fuel compacts enclosed in representative thicknesses of compact matrix and moderator graphite.

Past experience with uranium fuels will be utilized to the extent possible. Differences in behavior between uranium and actinide fuels would be the focus of these efforts (primary interest would be in the isotopes released and the relative amounts). Also, taking advantage of recent advances in computational simulation should allow the better design and optimization of these experiments. Accident condition testing will be performed in an out-of-pile furnace on the irradiated compacts, particles and fuel bodies. Reactivation of the fuel compacts to provide a source of short-lived isotopes will be necessary to characterize I and Te releases, unless these tests can be performed "real-time" in a loop test or if conservative estimates are acceptable (e.g., by assuming the fractional releases of short-lived I and Te isotopes are the same as that of 10.7-yr Kr-85).

Table 6-9 summarizes the elements in WBS 1.3, and Table 6-10 lists the planned in-pile experiments.

Table 6-9. WBS Elements for Fission Product and Actinide Transport

WBS 1.3: Fission Product and Actinide Transport	
WBS: 1.3.1	WBS:1.3.2
Compacted Fuel Particles	Fuel Bodies*
Chemical Evaluation	Irradiation Tests
Irradiation Tests	Accident Conditions
Accident Conditions	
* One or two compacts enclosed in moderator graphite	

Table 6-10. In-pile Tests in WBS 1.3

WBS 1.3 IN-PILE TEST SUMMARY			
WBS	Capsule	Contents	Conditions
1.3.1	FP-1	DTF DF particles in compacts	700-1300°C 40-80% FIMA
	FP-2	DTF TF particles and DF* particles in compacts	700-1300°C 40-80% FIMA
1.3.2	FP-3	DTF DF fuel bodies	700-1300°C 40-80% FIMA
	FP-4	MB TF/DF* particles	700-1300°C 40-80% FIMA
*DF particles are required in TF capsules for thermal performance.			

6.3.1 Radionuclide Retention in Fuel Compacts

6.3.1.1 Chemical Evaluation of Fission Products in DF and TF

A numerical procedure will be established to determine the quantity and chemical forms of fission products and actinides in the DF kernel as a function of burnup and temperature. Starting with a given kernel stoichiometry, the oxygen potential of the kernel will be determined as a function of burnup and temperature using thermodynamic and experimental data. The oxygen potentials will be used to determine the chemical forms of the fission products. A range of kernel compositions, both those containing PuO_{2-x} and oxycarbides, will be evaluated. Thermodynamic data and analysis will also be used to determine the physical forms of the fission products and actinides. In particular, the potential for increased internal pressure from CO generation is important, and the amount of potentially corrosive products that can migrate to the coatings will be estimated. Thermodynamic predictions will be compared with PIE data from capsule irradiations.

A similar procedure will be established for TF particles. A range of kernel compositions will be evaluated, as well as the effect of ZrC or other possible oxygen getters on the chemical forms of the fission products and on CO generation. Thermodynamic predictions for candidate kernel compositions will be performed and compared with PIE data.

6.3.1.2 Retention during Normal Operation

6.3.1.2.1 DF Compacts

Samples with the reference DF kernel composition will be irradiated in a 4-cell capsule (Capsule FP-1).[3] About 5,000 particles from a development fabrication run will be taken. A portion will be characterized for as-manufactured failed fuel fraction and actinide contamination level, leaving about 1,000 particles per irradiation cell. These particles will be formed into compacts using the same methods as for fabricating fuel compacts. The number of compacts will be chosen to satisfy a range of neutronic and thermal conditions. A helium sweep gas with appropriate levels of CO, CO_2, H_2O, and CH_4 will control the atmosphere of each cell. The sweep gas will be monitored on-line to determine fission gas (krypton and xenon) release from each cell as a function of burnup and temperature.

To provide a known fission product source, laser failed (LF), “designed-to-fail” (DTF)[4] or missing-buffer (MB) particles will be seeded in the compacts. The quantity of LF, DTF or MB particles will be approximately 10 times the expected number of normally failed particles so that the fission product source is quantifiable and the releases measurable; the fraction is expected to be in the 0.1–1.0% range. Based upon previous investigations with uranium fuels, the releases from LF or DTF particles are judged to be comparable to releases from actual failed particles, and the releases from MB particles should be prototypical since the failure of particles with missing or undersized buffers is expected to be an important source of in-reactor failure. Table 6-11 details the conditions in Capsule FP-1.

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[3] As for the Fuel Materials irradiations (Section 3.2), the planning basis here was a 4-cell capsule. The AGR program is planning to utilize a multi-cell capsule design with up to six independently purged and operated cells. If this design proves viable, it will be considered for use here as well. In that event, a greater range of test conditions could be investigated.

[4] “Designed-to-fail” (DTF) particles, as used here, are standard DF or TF kernels encapsulated by a 10-15 μm pyrocarbon seal coat; such DTF particles have been shown to fail rapidly under neutron irradiation, providing a well defined fission product source.

Table 6-11. Capsule FP-1 Conditions

CELL	TEST ARTICLE	TEMPERATURE (°C)	EOL FIMA[*] (%)
1	DF particles (DTF)	700	40-80%
2	DF particles (DTF)	900	40-80%
3	DF particles (DTF)	1100	40-80%
4	DF particles (DTF)	1300	40-80%

[*] Burnup consistent with capsule neutronics and temperature control

When preparing the test specification, the temperatures and burnups in Table 6-11 will be reviewed so that the most useful data is generated for model development when the actual reactor operating conditions are known and early information on DF performance is available.

Test cells in this capsule, FP-1 (and in the follow-on FP-2 and FP-3 capsules) should be operated at nearly constant temperature so that differential data for model development can be obtained for the design codes. Later validation tests in capsule FP-4 will be operated in a manner that approximates the expected temperature history of the reactor (to the extent practical).

The capsule internal components and the capsule temperature gradients will be designed to collect the released fission product metals on special deposition surfaces, to the extent possible. This will assure that the disassembly of the capsule can proceed in a straightforward manner with minimal handling and potential for contamination from hot cell sources. In any case, the capsules must be designed so that an accurate radionuclide mass balance can be obtained for each individual cell.

PIE activities will be selected to acquire a maximum amount of information from the irradiations. Gamma-, beta- and alpha spectroscopy and radiochemical analyses of cell surfaces and components will supply information on the total fission metal release during the irradiation. Acid leaching or washing of components will probably be necessary as well. The goal will be to obtain isotope-by-isotope mass balance as a function of failed fuel fraction.

6.3.1.2.2 TF Compacts

Capsule FP-2 will be designed to determine the RN release characteristics of TF particles in the same manner as DF particles in FP-1. However, DF particles will also be included in each cell of the capsule, as required, to generate the requisite fission power to maintain the required thermal conditions. The inclusion of DF particles in FP-2 may require that the “seeding” by LF or DTF TF particles be higher than in FP-1 so that the results are not overwhelmed by contributions by any DF defective particles. Table 6-12 details the capsule conditions for FP-2. As in FP-1, the temperatures and burnups should be reviewed when the test specification is prepared. This capsule will also be operated at constant temperature to collect differential data, and the PIE will proceed in a similar manner.

Table 6-12. Capsule FP-2 Conditions

CELL	TEST ARTICLE*	TEMPERATURE (°C)	EOL FIMA** (%)
1	TF particles (DTF)	700	40-80%
2	TF particles (DTF)	900	40-80%
3	TF particles (DTF)	1100	40-80%
4	TF particles (DTF)	1300	40-80%

*DF particles required for thermal performance

**Burnup consistent with capsule neutronics and temperature control

6.3.1.3 Retention under Accident Conditions

6.3.1.3.1 Driver Fuel

Irradiated compacts from capsule FP-1 will be subjected to accident temperatures in the Core Conduction Cooldown Test Facility (CCCTF), an existing furnace located in a hot cell which is specifically designed for this purpose. Each batch of irradiated DF particles will be held at 1400, 1500, and 1650°C for a period of 100 to 1000 hours depending on the particular data needed and the release behavior observed. A total of up to 12 such tests may be conducted, depending on the number of compacts available and on modeling needs. The maximum expected reactor accident temperature is expected to be no more than 1600°C; however, it is possible that tests higher than 1650°C may be conducted if thermal decomposition data is required, or if an advanced coating like ZrC is to be tested.

A cost-effective number of particles (after some of the compacts have been deconsolidated) may be placed in individually numbered holders so that their identity can be maintained throughout the test. If failures do occur, this individual tracking of particles will allow the failed particles to be examined on a one-on-one basis to determine the failure mechanism(s).

A furnace purge gas will provide control of the atmosphere and the means for continuous monitoring of fission gas release during the heating. There is provision for removable and replaceable deposition surfaces to monitor time-dependent fission metal release during the test. Examinations after the postirradiation heating will provide data similar to that collected in the irradiation tests: failed fuel fraction by the IMGA facility or leach-burn-leach, metallographic examinations of the kernel and coatings, and micro scale examination by the SEM/microprobe.

Since the releases of I and Te must be characterized, reactivation of one or more compacts or fuel bodies will be required and rapid transport of the fuel from the reactor to the test facility will be necessary. A likely list of the isotopes of interest in this case includes Ag-110m, Cs-134, Cs-137, Kr-85m, Kr-85, Kr-88, I-133, Sr-90, Xe-133, Xe-138, and Te-132.

6.3.1.3.2 Transmutation Fuel

Irradiated compacts from capsule FP-2 will be subjected to accident temperatures in the CCCTF. It is anticipated that the test matrix for heating Transmutation Fuel will be analogous to that for Driver Fuel outlined in Section 6.3.1.3.1.

6.3.2 Radionuclide Retention in “Fuel Bodies”

As discussed in Section 2, the radionuclide containment system for an HTGR is comprised of multiple barriers to limit radionuclide release from the core to the environment to insignificant levels during normal operation and postulated accidents. The first three release barriers - kernels, coatings, and matrix/graphite - are located within the reactor core. In this FDP, the test article collectively representing these three release barriers is referred to a “fuel body.” It consists of DF and/or TF compacts contained within a graphite structure that approximates the unit cell of a prismatic fuel element. Such test articles are used in this program to generate two distinct types of experimental data: (1) differential data on the radionuclide retention characteristics of the fuel-compact matrix and fuel-element graphite, and (2) integral release data from the entire assembly representing RN release from a fuel-element. The former data will be used to improve component models, and the latter data will be used to validate the design methods used to predict radionuclide releases from the full core.

6.3.2.1 Retention in “Fuel Bodies” during Normal Operation

6.3.2.1.1 Retention by Matrix and Graphite

While the coatings on the individual particles are the primary barrier to radionuclide release from the reactor core, the fuel-compact matrix and the fuel-element graphite are essential release barriers for those radionuclides that are not fully contained by intact fuel particles. Specifically, the matrix and graphite can significantly attenuate the release of fission metals, such as Cs and Sr, and preclude the release of actinides from the core during normal operation and during core heatup transients. 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 (Section 3.3).

Capsule FP-3 is designed to determine the radionuclide retention characteristics of matrix and graphite under DB-MHR core service conditions, especially for the high fast fluences for which there are no existing data.

The test specimens for capsule FP-3, referred to here as a "fuel body," will consist of DF compacts containing DF and TF "designed-to-fail" particles surrounded by three concentric annuli: first, an annulus of compact-matrix material, then an annulus of fuel-element graphite (the replacement for H-451), and finally an annular char-impregnated graphite to serve as a radionuclide sink.

Four fuel bodies will be irradiated in capsule FP-3 under test conditions given in Table 6-13. The cells in this capsule will operate isothermally to the extent possible to provide differential data for model improvement. When preparing the test specification, the design temperatures and burnups should be reviewed as well.

Table 6-13. Capsule FP-3 Conditions

CELL	TEST ARTICLE	TEMPERATURE (°C)	EOL FIMA* (%)
1	DF body (DTF)	700	40-80%
2	DF body (DTF)	900	40-80%
3	DF body (DTF)	1100	40-80%
4	DF body (DTF)	1300	40-80%
*Burnup consistent with capsule neutronics and temperature control			

The purge gas through each cell will continuously be monitored for fission gas release to determine the time-history of DTF particle failure. The primary experimental data will be obtained during PIE and will consist of radionuclide inventories in the fuel body components and of RN concentration profiles in the matrix and graphite annuli. Selected intact fuel bodies will be reserved for accident testing.

6.3.2.1.2 Integral Validation Data

In addition to the differential data to be obtained from capsules FP-1, -2 and –3 for model development, independent integral RN transport data are needed to provide the basis for validating the analytical methods used to predict radionuclide release from the DB-MHR core. Capsule FP-4 is designed to provide these integral validation data.

The test specimens in capsule FP-4 will have the same configuration as those in capsules FP-1 and FP-2, but each test cell will contain two types of compacts: one type will consist of DF particles seeded with missing-buffer DF particles, and the other will consist of DF particles seeded with missing-buffer TF particles. In other words, it will be as representative of the core as practical. In contrast to the first three capsules, capsule FP-4 will operate with a variable temperature history which approximates the time-temperature history in the reactor core to the extent practicable in an irradiation capsule. The nominal service conditions are given in Table 6-14. The capsule design and service conditions will be carefully reviewed when the test specification is prepared.

Table 6-14. Capsule FP-4 Conditions

CELL	**TEST ARTICLE**[*]	**TEMPERATURE (°C)**	**EOL FIMA**[**] **(%)**
1	TF/DF* compacts (MB)	700	40-80%
2	TF/DF* compacts (MB)	900	40-80%
3	TF/DF* compacts (MB)	1100	40-80%
4	TF/DF* compacts (MB)	1300	40-80%

[*] DF particles required for thermal performance

[**] Burnup consistent with capsule neutronics and temperature control

Fission gas release data will be obtained continuously during the irradiation as a function of burnup and temperature. Integral fission metal and actinide release data will be obtained during the PIE using the techniques outlined above for capsules FP-1 and FP-2. Selected fuel compacts and/or whole fuel bodies will be reserved for accident testing.

6.3.2.2 Radionuclide Retention under Accident Conditions

6.3.2.2.1 Retention by Fuel Matrix and Graphite

Irradiated DF fuel bodies from capsule FP-3 will be subjected to accident condition testing in the CCCTF. Each fuel body will be held at 1400, 1500, and 1650°C for a period of 100 to 1000 hours depending on the particular data needed. A furnace purge gas will provide control of the atmosphere and means for continuous monitoring of fission gas release. There is provision for removable and replaceable deposition surfaces to monitor fission metal release during the test. Radionuclide inventories in the fuel body components and concentration profiles in the matrix and graphite annuli will be determined by conventional methods after the heating is complete. Particular emphasis will be placed upon quantifying any measurable releases of actinides from the intact fuel bodies during the heating, especially at the highest test temperature.

6.3.2.2.2 Integral Validation Data

Irradiated DF and TF fuel compacts from FP-4 will be subjected to accident condition testing in the CCCTF to obtain integral release data for design methods validation. Each fuel compact will be subjected to a variable time-temperature history representative of that experienced in the reactor core; it is anticipated that a peak temperature of 1600°C will be specified for most if not all of the tests. The detailed test matrix will be finalized when the test specification is prepared.

The furnace purge gas will provide control of the atmosphere and means for continuous monitoring of fission gas release. There is provision for removable and replaceable deposition surfaces to monitor fission metal release during the test. Posttest examinations will provide data similar as in the irradiation tests: failed fuel fraction by the IMGA facility, metallographic examinations of the kernel and coatings, and micro scale examination by the SEM/microprobe. Since the releases of I and Te isotopes must be characterized, reactivation of fuel compacts and/or whole fuel bodies will be necessary.

6.4 Spent Fuel Disposal (WBS 1.4)

This section describes the proposed confirmatory tests and analyses to qualify DB-MHR spent fuel as a suitable waste form for permanent disposal in a geologic repository and to assess the transport of radionuclides from spent fuel elements in the repository to the environment. The proposed tests are summarized and prioritized in Table 6-15.

In the process of performing the previous assessments of HTGR spent fuel in a repository environment which were described in Section 3.4, it became apparent that certain additional experimental data, primarily related to the long-term stability of coated-particle fuel, would serve to increase the confidence in the predictions; consequently, confirmatory test and analysis plans defining experimental programs to generate such data were prepared for the PC-MHR in 1995 (Hanson 1995) and for the commercial GT-MHR in 2002 (Hanson 2002).

The spent fuel DDNs, which included herein as Appendix A, are largely generic, or the differences for different fuel cycles, are obvious (e.g., different kernel compositions); hence, the plans prepared for PC-MHR and GT-MHR provided a reliable basis for identifying the comparable spent fuel DDNs for the DB-MHR and for defining the attendant test programs. The results are summarized in this section. The generic spent fuel DDNs (e.g., C-14 leaching rates from graphite), or portions thereof, could well be satisfied by other HTGR programs depending upon funding priorities and schedules. The DB-MHR specific DDNs (e.g., actinide leaching rates from DF and TF kernels) will have to be addressed by this program as described below.

The purpose of this section is to provide an overview of the proposed work. The various DDNs are grouped so that they can be most efficiently addressed in common test programs (e.g., RTW.04.01, Radionuclide Leaching Rates from DF Kernels, and RTW.04.02, Radionuclide Leaching Rates from TF Kernels are combined). As the actual work is authorized, more detailed planning documents would be prepared to define the workscope more completely and, in particular, more quantitatively; e.g., for experimental work, both test specifications and test procedures would be issued, before initiating the actual testing. Nevertheless, even at this early stage, it is possible to define, in a generic way, the anticipated experimental conditions for the proposed testing. A preliminary set of test articles and test conditions is presented in Table 6-16.

Table 6-15. Confirmatory Tests to Support Spent Fuel Disposal

SECTION	CONFIRMATORY TEST	DDNs	GENERIC DESCRIPTION	PRIORITY*
6.4.1.1	Long-term mechanical integrity of TRISO coatings	RTW.04.03	PIE of historical irradiated TRISO particles, TBD	H
6.4.2.1.1	PyC coating oxidation rates (air)	RTW.04.04	Accelerated (temperature) tests with TRISO particles	L
6.4.2.1.2	SiC coating oxidation rates (air)	RTW.04.05	Accelerated tests with TRISO particles (OPyC removed)	M
6.4.2.1.3	Resin matrix oxidation rates (air)	RTW.04.06	Accelerated lab tests	L
6.4.2.1.4	Fuel-element graphite oxidation rates (air)	RTW.04.07	Accelerated lab tests with unirradiated/irradiated samples	L
6.4.2.1.5	Graphite noncombustibility demonstration	RTW.04.08	Combustibility testing per ASTM standard (and future standards if necessary)	H
6.4.2.2.1	PyC coating corrosion rates (groundwater)	RTW.04.09	Accelerated and real-time tests with spectrum of water chemistries	M
6.4.2.2.2	SiC coating corrosion rates (groundwater)	RTW.04.10	Accelerated and real-time tests with spectrum of water chemistries	H
6.4.2.2.3	Resin matrix corrosion rates (groundwater)	RTW.04.11	Accelerated and real-time tests with spectrum of water chemistries	M
6.4.2.2.4	H-451 graphite corrosion rates (groundwater)	RTW.04.12	Accelerated and real-time tests with unirradiated/irradiated samples	M
6.4.3.1	Impurity content of matrix and graphite	RTW.04.13 RTW.04.15	N-14 content in archival graphite; C-14 content in irradiated H-451 graphite	H
6.4.3.2.1	Radionuclide leaching rates from kernels	RTW.04.01 RTW.04.02	Accelerated tests with failed PuO_x particles from FTE-13 Accelerated tests with failed DF and TF particles	L
6.4.3.2.2	Radionuclide leaching rates from graphite	RTW.04.14 RTW.04.16	Accelerated lab tests with irradiated H-451 samples	H

*Relative Priority: H = High; M = Medium; L = Low

Table 6-16. Generic Test Conditions for Experiments

TEST ARTICLES	
Fuel particles	DF/TF fuel compacts; loose DF/TF particles
Matrix type	Thermosetting resin
Neutron poison	PyC-coated B_4C PyC-coated Er_2O_3
Graphite body	“H-451”; unit cell of prismatic element
NORMAL REPOSITORY ENVIRONMENT	
Fuel burnup range	[0 to 85]% FIMA DF/[0 to 85]% FIMA TF
Peak fast fluence	≤ 8 x 10^{25} n/m^2 (E > 29 fJ)
Residence time	$\leq 10^6$ years
Repository pressure	1 atm
Fuel element temperatures	[20^o to 250 ^{o}C]
Environment (Air)	[$\leq$0.01 atm H_2O] [0.21 atm O_2] [0.79 atm N_2]
Expected exposed fuel fraction	[10^{-5} to 10^{-3}]
OFF-NORMAL CONDITIONS (GROUNDWATER INGRESS)	
Groundwater chemistry	Yucca Mountain, TBD
Vapor-phase environment	$O_2/N_2/H_2O/CO/H_2$
Vapor-phase impurity levels	[0 to 1.0] atm H_2O
Fuel-element temperatures	[20 - 100^{o}C]
Groundwater contact time	TBD
Repository pressure	$\geq$1 atm
Expected failure fraction	[10^{-4} to 10^{-2}]

This section describes the acquisition of additional test data to increase the confidence that DB-MHR spent fuel will be accepted as a suitable waste form by regulatory authorities.[5] Previous assessments have shown that the existing data, when applied conservatively with sufficient margin to account for estimated uncertainties (as was done in the PC-MHR and GT-MHR repository analyses) predict compliance with anticipated regulatory requirements. It is expected that the scope of any confirmatory testing program will be negotiated between the regulatory authorities and the applicant as a condition of obtaining the requisite approvals for GT-MHR spent fuel disposal.

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[5] It is recognized that predictions of fuel performance and radionuclide transport on geologic time scales will always be inherently uncertain, regardless of the completeness of the “short-term” data base upon which these predictions are necessarily predicated.

The tests proposed in this section are relatively simple, inexpensive, and are a small fraction of the entire fuel plan cost. The experiments are laboratory-scale and require test facilities, instrumentation, and support services that are typically found in a modern, well-equipped radiochemistry laboratory. Tests with irradiated material will require a hot cell. The experiments are not complicated but generally of long-term duration (in some cases, multiple years). The most sophisticated apparatus would likely be conventional high-pressure autoclaves for accelerated corrosion and leaching tests with (simulated) liquid groundwater. Certain of the test programs could begin early, essentially immediately upon availability of funding, because they would utilize existing samples of unirradiated and irradiated TRISO-coated fuel particles and H-451 graphite which are available at ORNL.

6.4.1 Spent Fuel Performance

6.4.1.1 Long-Term Mechanical Integrity of Stressed TRISO Coatings

The TRISO coatings on HTGR spent fuel particles offer the potential for long-term (e.g., 10^4 years or more) containment of radionuclides, well beyond that provided by the waste package (waste canister, backfill, etc.). Confirmation of the long-term integrity of TRISO coatings in a repository environment is needed to support repository performance assessments for comparison with regulatory criteria.

The long term, near-field containment of radionuclides by DB-MHR spent fuel depends in large part upon the capability of the TRISO coatings to withstand for long times the stresses resulting from internal gas pressure (fission gas and CO pressure plus helium pressure buildup due to alpha decay). It is believed that TRISO coatings will maintain their mechanical strength indefinitely, assuming that the coatings are not chemically or physically degraded by external events (e.g., by groundwater corrosion, volcanism, etc.). In evaluating the long-term mechanical performance of TRISO-coated fuel particles, it is important to consider the multiple-layer coating system as a composite structure. For example, in a properly designed TRISO particle, the OPyC and IPyC coatings serve to keep the SiC coating in compression even at peak burnups, and SiC coatings are stronger in compression than in tension.

This capability has been evaluated for PC-MHR and commercial GT-MHR spent fuel using static calculations of SiC hoop stress and measurements of SiC strength fitted to a Weibull distribution. These methods and data have been used previously to evaluate the ability of TRISO coatings to withstand internal pressure for the relatively short times relevant to fuel irradiation (typically ~3 years) and to temporary storage (e.g., up to ~30 years). It is believed that these calculational methods and data are also suitable for determining the integrity of TRISO coatings for the much longer times associated with permanent fuel disposal in a repository, but this needs to be confirmed to the extent practicable.

Data are needed which would reduce the present uncertainty in the calculation of the capability of TRISO coatings to withstand the level of internal gas pressure expected to develop in DB-MHR spent fuel over the time frames important to geologic disposal. The crush strength should be measured for coated particles that have been stored for different time periods, to determine the effect of storage time upon the crush strength. The intent is to generate additional empirical data to determine if the strength of TRISO coatings degrades measurably with time.

High-burnup Pu particles recovered from fuel test element FTE-13 (Miller 1985), which was irradiated in the Peach Bottom-1 reactor and which is currently in storage, would be an ideal early source of test specimens. Such data may enable a correlation to be developed which would allow extrapolation of the crush strength to long time periods.

The experimental approach would be to perform tests to acquire the data described above while at the same time pursuing geologic analog data (e.g., moissanite). For example, it should be determined whether ancient moissanite formations are largely coherent or heavily faulted, etc. Reliance geologic analogs is not considered sufficient by itself because of the expected physical and chemical differences between naturally occurring SiC deposits and SiC coatings produced by chemical vapor deposition.

6.4.2 Corrosion of Fuel-Element Components

The long term, near-field containment of radionuclides by DB-MHR spent fuel depends in large part on the capability of the fuel particle coatings to withstand for long times the stresses resulting from internal gas pressure (end-of-irradiation fission gas and CO pressure plus helium pressure due to alpha decay). The containment capabilities of the TRISO coating system may be compromised by corrosion of the coatings by air or by groundwater which may intrude into the repository.

6.4.2.1 Oxidation by Air

The air oxidation rates of the PyC and SiC coatings at repository temperatures are expected to be negligibly small based upon extrapolation of the measured rates at the highly elevated temperatures characteristic of normal reactor operation and postulated accidents. Nevertheless, experimental confirmation is desirable by measuring the oxidation rates at intermediate temperatures at which the rates can still be reliably characterized in long-term tests (e.g., months in duration).

6.4.2.1.1 PyC Oxidation

In the performance assessment for the PC-MHR, it was assumed that the load resulting from the internal gas pressure is borne only by the SiC coating layer; contributions to the coating system structural integrity provided by the OPyC coating were neglected; for the commercial GT-MHR, two cases were considered: (1) the OPyC coating was neglected, and (2) the OPyC coating was assumed to be sacrificially corroded and to protect the SiC for the first 400,000 years.

Neglecting the OPyC is conservative, because, for a long period of time, depending in part on the rate of OPyC oxidation by the repository environment, the OPyC layer will: (1) carry part of the load arising from internal gas pressure and serve to keep the SiC layer in compression, and (2) protect the SiC layer from oxidation from the repository environment. Including these effects in particle performance models will make the DB-MHR spent fuel performance assessment more realistic; however, this approach requires a better knowledge of the rate of oxidation of the PyC layer in air at expected repository temperatures.

More fundamentally, taking credit for the OPyC coating as a structural member and a release barrier is consistent with the defense-in-depth philosophy which is particularly important in the present context since it will be impractical to predict with high confidence the performance of any particular material over geologic times.

Additional data are needed by which to estimate the oxidation rate of PyC by air at repository temperatures to allow the contribution of the OPyC layer to be included in the evaluation of the long term integrity of TRISO coatings. To generate these data would require accelerated tests of PyC oxidation in air conducted at temperatures intermediate between reactor operating temperatures and repository temperatures. Tests would be conducted over a range of temperatures for extended periods of time. The data would need to be sufficient to enable extrapolation of the oxidation rates to long times (e.g., thousands of years). Some measurements would need to be made with irradiated TRISO particles to determine whether or not there are any irradiation effect.

6.4.2.1.2 SiC Oxidation

The long term, near-field containment of radionuclides by HTGR spent fuel depends in part on the capability of the fuel particle SiC coatings to withstand oxidation by air. The commercial GT-MHR spent fuel performance assessment included an estimate for the expected oxidation of the SiC coatings by air at repository temperatures (the expectation is negligible SiC oxidation, even for geologic time frames). This estimate is based on an extrapolation using a parabolic rate law of SiC oxidation rates measured at temperatures much higher than those expected in a repository; however, the accuracy of this estimate is uncertain.

Additional data are needed on the oxidation of SiC in air at temperatures closer to those expected in a repository to verify the estimate of SiC oxidation to be used in the DB-MHR spent fuel performance assessment. To generate these data will require accelerated tests of SiC oxidation on TRISO particles from which the OPyC layer has been removed and conducted at temperatures intermediate between reactor operating temperatures and repository temperatures. The data would need to be sufficient to enable extrapolation of the measured oxidation rates to long times (e.g. hundreds or thousands of years). Some of the measurements should be made with irradiated TRISO particles with the OPyC layer removed to determine whether or not there are irradiation effects.

6.4.2.1.3 Matrix Oxidation

Although not considered in the previous HTGR spent fuel assessments, it is believed that the compact thermosetting resin matrix will contribute to the long-term radionuclide retention of the fuel by providing a fuel form which is mechanically stable for long times and by serving as an important sorptive sink for condensible radionuclides. For these effects to be evaluated quantitatively, knowledge is needed of the long-term rate of oxidation by air of resin matrix at temperatures expected in the repository.

The resin matrix is also a potential source of C-14 release to the environment. An accurate estimate of the long-term rate of oxidation by air of GT-MHR resin matrix at repository temperatures is also needed to assess the potential for C-14 release in the form ^{14}CO and/or $^{14}CO_2$.

Additional data are needed on the oxidation rate of resin matrix in air so that it can be included in the evaluation of the long term capability of DB-MHR spent fuel to retain radionuclides. Generating these data will require accelerated tests of resin matrix oxidation in air conducted at temperatures intermediate between reactor operating temperatures and the expected repository temperatures. The data would need to be sufficient to enable extrapolation of the oxidation rates to long times (e.g., thousands of years). Some of the measurements should be made with irradiated matrix recovered from irradiated fuel compacts to determine whether or not there are irradiation effects, including possible catalysis by metallic fission products sorbed on the matrix.

6.4.2.1.4 Graphite Oxidation

The fuel block, which is made from “H-451” graphite,[6] will also contribute to the long-term radionuclide retention of the fuel by providing an overall fuel form which is mechanically stable for long times and by serving as a sorptive sink and a diffusion barrier for condensible radionuclides released from the fuel compacts in the reactor and in the repository. For these effects to be evaluated, information is needed regarding the long-term oxidation rate of H-451 graphite at repository temperatures. The H-451 graphite is a potential source of release of C-14 and other sorbed radionuclides to the environment. An accurate estimate of the long-term rate of oxidation of H-451 graphite at repository temperatures is also required to accurately assess potential C-14 release in the form ^{14}CO and/or $^{14}CO_2$.

Additional data are needed to allow a more accurate estimate of the oxidation rate of H-451 graphite in air at repository temperatures. To generate these data will require accelerated tests of H-451 graphite oxidation in air conducted at temperatures near those expected in the repository. The data would need to be sufficient to enable extrapolation of the oxidation rates to long times (e.g. thousands of years). Some of the measurements would need to be made with irradiated H-451 graphite to determine whether there are irradiation effects, including possible catalysis by metallic fission products sorbed on the graphite.

6.4.2.1.5 Graphite Noncombustibility Demonstration

A current generic requirement for waste disposal in a geologic repository, which is given in 10CFR60, is that the waste form needs to be “noncombustible;” however, no specific quantitative definition of combustibility is provided, so an alternative standard must be selected. A common industry standard used to assess the combustibility of structural materials is ASTM E-136. The available (albeit circumstantial) evidence indicates that H-451 is clearly “noncombustible” when judged by this ASTM standard.

Additional data are needed to confirm the initial determination that H-451 graphite and fuel-compact matrix are noncombustible according to the criteria outlined in ASTM Standard E-136. The most direct way to provide this confirmation would be to test H-451 and matrix specimens

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[6] It is assumed that the reference fuel-element graphite for the DB-MHR will be the functional equivalent of H-451.

in accordance with the ASTM standard; both unirradiated and irradiated specimens should be tested. Additional testing may ultimately be required if regulatory authorities were to adopt a different quantitative definition of combustibility.

6.4.2.2 Corrosion by Groundwater

Repository flooding by groundwater has the potential to be a major disruptive event for a geologic spent fuel repository because it could significantly degrade performance of the repository's natural and engineered barriers and could provide a direct release pathway to the environment. For Yucca Mountain the DOE has concluded that repository flooding is not scientifically credible, primarily because of the geologic record of the site. Nevertheless, it is quite possible that the NRC and/or intervenors will raise the issue of groundwater flooding during the licensing process so it seems appropriate to characterize the performance of the final waste package under these conditions. Moreover, based upon the existing data, it is evident that HTGR spent fuel is far more resistant to groundwater corrosion than the reference LWR waste package which may be an important advantage.

6.4.2.2.1 OPyC Corrosion

The long-term containment of spent TRISO-coated fuel will depend largely on the performance of the PyC and SiC coatings. The PyC coatings may be corroded by groundwater during off-normal repository environment conditions (i.e., groundwater ingress). Corrosion of the outer PyC coating is a potentially important mechanism that can affect long-term fuel performance. Moreover, as stated previously , taking credit for the OPyC coating as a structural member and a release barrier is consistent with the defense-in-depth philosophy which is particularly important in the present context since it will be impractical to predict with high confidence the performance of any particular material over geologic times. Therefore, corrosion characteristics of the OPyC coating must be quantified for the off-normal conditions.

Correlations will be developed describing the corrosion rates for the OPyC coatings of irradiated, TRISO-coated fuel particles under groundwater-ingress conditions; irradiated TRISO-coated particles from a number of HTGR fuel irradiation capsules, which are currently in long-term storage at ORNL, would be an ideal early source of test specimens. Both real-time tests at ambient temperature and pressure and accelerated tests in high-pressure autoclaves will be performed with a range of water chemistries and temperatures.

6.4.2.2.2 SiC Corrosion

Following the dissolution of the OPyC coating by groundwater interaction, the SiC coatings may be subsequently corroded. Corrosion of the SiC is potentially the controlling mechanism in determining long-term fuel performance and near-field source terms. Therefore, corrosion characteristics of the SiC coating must be quantified.

Correlations will be developed describing the corrosion rates of SiC coatings of irradiated TRISO-coated fuel particles under groundwater-ingress conditions; irradiated TRISO-coated particles from a number of HTGR fuel irradiation capsules, which are currently in long-term storage at ORNL, would be an ideal early source of test specimens. Both real-time tests at ambient temperature and pressure and accelerated tests in high-pressure autoclaves will be performed with a range of water chemistries and temperatures.

6.4.2.2.3 Matrix Corrosion

The carbonaceous matrix materials used as binders in the fuel compacts, lumped burnable-poison (LBP) compacts, and the reserve shutdown control (RSC) pellets consist of finely divided graphite flakes bonded together with residual carbon from carbonized, thermosetting resin binders. The matrix may be corroded by groundwater during off-normal repository environment conditions. The corrosion characteristics of the resin-matrix material should be quantified for the groundwater ingress conditions.

Correlations will be developed describing the corrosion rates for resin-matrix materials under groundwater-ingress conditions. To characterize the reaction kinetics, the reaction rate needs to be determined as a function of water chemistry, temperature, and time. The effect of partial matrix burnoff on both the mass transfer processes and the reaction kinetics will also be quantified. Both real-time tests at ambient temperature and pressure and accelerated tests in high-pressure autoclaves will be performed with a range of water chemistries and temperatures.

6.4.2.2.4 Graphite Corrosion

The DB-MHR fuel compacts are contained in H-451 graphite blocks. During postulated groundwater ingress, the fuel-element graphite will be corroded. Corrosion of the fuel-element graphite is a potentially important mechanism that can affect long-term fuel performance in the repository environment, including the mechanical stability of the waste form and is a potentially important source of C-14 release from the repository; therefore, corrosion characteristics of the H-451 graphite will be quantified.

Data will be obtained for the corrosion rates of the H-451 graphite during groundwater-ingress conditions. To obtain these data and subsequently to develop corrosion rate correlations, accelerated tests in high-pressure autoclaves will be performed as a function of water chemistry, temperature, and time. The variation in water chemistry should cover the range from the distilled water to water similar in chemistry to Yucca Mountain site groundwater. Some tests need to be performed on irradiated H-451 to investigate possible irradiation effects. (The available data imply that the effect will be unimportant.)

6.4.3 Radionuclide Transport

6.4.3.1 Radionuclide Inventories in Spent Fuel

In general, the radionuclide inventories in HTGR spent fuel can be calculated with sufficient accuracy with nuclear diffusion depletion codes, such GARGOYLE or ORIGIN2; however, with DF and TF fuel, nuclear reactions involving higher actinides are more important than with conventional U and Th fuel cycles which introduces additional uncertainties. As a result, there is significant uncertainty regarding the total production of helium atoms by alpha decay over geologic times; therefore, DDNs RTW.02.04 and 02.05 were identified to better characterize the gas production in DB-MHR spent fuel.

There are considerable uncertainties in the total inventories of the radionuclides produced by neutron reactions with the trace impurities in the core structural materials. A major reason for this uncertainty is that the concentrations of these impurities are not sufficiently quantified which is addressed in this subsection.

6.4.3.1.1 C-14 Content of Matrix and Graphite

Carbon-14 is an activation product, which is produced in HTGR fuel elements primarily through the neutron activation of N-14, via an (n,p) reaction, which exists as an impurity in graphite. Nitrogen-14 is introduced during the manufacture of the fuel compacts and the graphite fuel blocks. The other mode of C-14 production is by the activation of C-13, which is a rare isotope (1.1% abundance) that is naturally present in graphite. Carbon-14 is one of the key radionuclides of concern for release by groundwater leaching. Its relatively long half-life (5730 years) and high mobility in the geosphere make it a potential long-term health hazard. To calculate C-14 release rates, the inventories of C-14 in the spent fuel-element graphite and fuel-compact matrix must be quantified.

Data will be generated to permit reliable quantification of the C-14 content of irradiated H-451 graphite in compact matrix material. Tests will be performed to measure C-14 inventories in the H-451 graphite irradiated in the FSV reactor and/or HTGR fuel irradiation capsules. Data will also be obtained regarding the N-14 content in as-manufactured graphite from archival specimens. When a replacement graphite for H-451 has been manufactured, its N-14 content will be measured as well.

6.4.3.1.2 Chemical Impurities in H-451 Graphite

Nuclear-grade graphites are by necessity high-purity materials because many of the chemical impurities that are typically present in the feedstocks for synthetic graphites are neutron poisons, and some of them catalyze graphite corrosion reactions, thereby limiting the allowable service conditions for certain applications. Fuel-compact matrix materials derived from thermosetting resins are also typically high purity materials; petroleum pitch, which was the feedstock for FSV compact matrix, is a relatively impure material, but multiple cleaning steps, including high temperature gaseous HCl leaching, are used to remove these impurities from the matrix in the finished fuel compacts. Nevertheless, the fuel and graphite product specifications do allow low levels of impurities, and some of these residual impurities may be transmuted in the reactor to very long-lived radionuclides which can contribute significantly to predicted repository dose rates at long times. For example, in a British study of permanent disposal options for irradiated Magnox and AGR graphite core components, the dominant radionuclide for certain long-time scenarios was 300,000-yr Cl-36 which is produced by neutron activation of trace Cl impurities in the graphite (White 1984). Consequently, the trace chemical impurities permitted in DB-MHR core graphite and matrix materials need to be thoroughly characterized.

The chemical impurities in H-451 graphite and in fuel-compact matrix material will be fully characterized, and the spectrum of nuclear transmutation products that can be generated upon irradiation in an HTGR neutron spectrum will be determined by both direct measurement and analysis with appropriate nuclear depletion codes, such as GARGOYLE or ORIGEN2. Certain of these determinations will need be repeated once a new coke source for the future manufacture of H-451 equivalent graphite has been identified. The test program will include high-sensitivity gamma spectroscopy and radiochemistry of irradiated H-451 graphite specimens recovered from irradiated FSV fuel elements.

6.4.3.2 Radionuclide Leaching Rates

6.4.3.2.1 Radionuclide Leaching Rates from DF and TF Kernels

A small fraction of the TRISO coatings is expected to experience in-service and in-storage failure, resulting in exposed DF and TF kernels. In the event of a failure of the waste canister, designed to contain the DB-MHR spent fuel elements, a spent fuel element may be in contact with groundwater. In such case, the radionuclide release rates to the groundwater will depend upon the effective leaching rates. For the commercial GT-MHR performance assessment, it was assumed that the radionuclide inventory in the exposed fuel kernels would be leached in a relatively short time compared with the disposal time scale. This estimate is based upon the German data for leaching rates of low-burnup UO_2 kernel radionuclide inventories in brine. For DB-MHR fuel, relevant data are needed for leaching rates of kernel radionuclide inventories in groundwater.

Correlations will be developed describing the leaching rates for radionuclides in exposed DF and TF kernels under groundwater-ingress conditions. Data will be obtained from accelerated tests with failed DF and TF particles (i.e., particles with exposed kernels); high-burnup PuO_x particles from FTE-13, which are currently in long-term storage, would be an ideal early source of test specimens. Correlations will cover the range of fuel temperatures for normal and off-normal conditions and a range of groundwater chemistries. The tests will necessarily be accelerated, but they will be of sufficient duration and at sufficiently elevated temperatures so that statistically significant quantities of the key radionuclides, including Pu and other actinides, will be dissolved and detected in the water.

6.4.3.2.2 Radionuclide Leaching Rates from Matrix and Graphite

The radionuclides in a HTGR spent fuel element are effectively retained (>99.9 %) by the TRISO-coated fuel particles. Nevertheless, a small but important fraction of the radionuclides are found outside of the coated particles (as a result of as-manufactured heavy-metal contamination, failed particles, neutron activation, etc.) but sorbed on the fuel-compact matrix and graphite; these radionuclides are much more readily releasable from the fuel elements than those contained by intact TRISO coatings. In particular, during postulated groundwater ingress, these inventories can be released to groundwater by leaching of the graphite and matrix; however, the effective leaching rates may be limited by solubility limits and mass-transfer rates.[7] The fission product leaching rates from the irradiated H-451 fuel-element graphite, especially for C-14, will be quantified.

Data will be obtained for radionuclide leaching rates from irradiated H-451 graphite because the current data base does not include leaching rates from this graphite which is the DB-MHR

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[7] It is noteworthy that 10CFR60 imposes *annual* fractional release limits so that the leach rates over that time frame are of interest.

reference. At least some of the tests will be performed in high-pressure autoclaves to obtain liquid water corrosion data at temperatures above 100°C; other tests will be real-time tests at ambient temperatures and pressures. Radionuclides of interest include fission products, activation products, and actinides, especially Pu isotopes. Accelerated lab tests will be performed to measure leaching rates from the H-451 graphite irradiated in the FSV reactor and/or HTGR fuel irradiation capsules.

6.4.3.3 Radionuclide Retention by Overpack and Backfill

As discussed above, unprocessed HTGR spent fuel, including DB-MHR spent fuel, is judged to be a nearly ideal final waste form which is predicted to effectively retain radionuclides for geologic time periods. Because of the robustness and retention characteristics of the TRISO fuel particles and the fuel-element graphite, it should not be necessary to include additional engineered barriers such as overpack and drip shields in the waste package for HTGR spent fuel. Nevertheless, overpack/backfill could be used in conjunction with HTGR spent fuel to provide additional defense-in-depth, but retardation by the overpack/backfill could most likely be conservatively neglected when showing compliance with radionuclide retention requirements. Consequently, no DDNs are presented herein regarding radionuclide retention by such additional barriers or regarding radionuclide transport in the geosphere; in any case, such DDNs would be generic, and it is assumed that the ongoing international technology programs to qualify various candidate geologic repositories for the permanent disposal of spent LWR fuel would provide the requisite data in a timely manner.

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7. Facility Requirements for DB-MHR Fuel Development Plan

7.1 Introduction to Facility Requirements

The program has demands for unique facilities and equipment. The requirements are detailed in the subsequent sections along with discussion of facilities and capabilities that can be used to fulfill the requirements. Also indicated in the details is whether the capability or equipment currently exists. If it does not exist, it has been provided for in the estimates for this program.

Facility requirements for the development program include buildings and equipment for the following major subdivisions of the program:

1. Fuel fabrication process development,
2. Fuel characterization, testing, and test capsule preparation,
3. Fuel irradiation testing,
4. Post-irradiation examination and testing.

The requirements of the program will be addressed for each of these major subdivisions, along with an assessment of capability of existing facilities at ORNL and how they might be used in the program. A listing of equipment and facilities that are to be funded by the program is included in this section.

The isotopes to be handled in the program dictate the use of a variety of facilities designed to control the radiation and hazards of the materials being handled. The program will be processing and testing bare materials and components containing:

1. Uranium fuels requiring controlled ventilation and radiation control,
2. Plutonium-based fuels requiring glove boxes for operations, and
3. Transmutation fuels requiring shielded hot cells design for alpha-radiation containment.

In addition, the program will require hot cell facilities with alpha-radiation containment for post-irradiation examination and testing of fuel and for reprocessing development. Supporting services such as analytical chemistry, metallurgical services, and waste management are also required.

7.2 Description of Existing Facilities

Facilities satisfying many of the requirements for this program exist at ORNL and INEEL. Facilities would be selected from the following list. A description is given for each, along with an evaluation of what would be needed to equip each for the intended work.

Uranium fuels fabrication and characterization at ORNL:

- Metals and Ceramics Research and Development Laboratory, Building 4508
- Process Development Laboratory, Building 4501
- Robotics Laboratory, 7600 Complex
- Post Irradiation Examination Laboratory, Building 4501

DF fabrication and characterization at ORNL:

- Alpha Laboratories, Building 7920
- Alpha Laboratories, Building 7930
- Transuranium Research Laboratory, Building 5505
- Potential conversion of space in Building 3525

TF fabrication and characterization at ORNL:

- Hot Cells, Building 7930
- Hot Cells, Building 3525
- Transuranium Research Laboratory, Building 5505 (very small quantities)

Irradiation Testing:

- High Flux Isotope Reactor, Building 7910 (ORNL)
- Advanced Test Reactor (INEEL)

Post Irradiation Examination at ORNL:

- Post Irradiation Examination Facility, Building 3525

While all of these facilities exist, most of them will need to be equipped with the special equipment required for fabrication of fuels. Some characterization equipment exists, but the facilities must be enhanced for the fuels applications. Thus, in the program plan, significant costs are allocated in equipment cost category and also in direct costs for installing the capability of doing the work of this program plan

ORNL essentially dismantled equipment for fabricating fuel in the early 1980s, with the demise of its comprehensive fuel cycle and fuel development programs. However, many of the facilities capable of accommodating the fabrication processes still remain at ORNL and can be commissioned to do the work of this program plan without large expense. Some new fuel fabrication process equipment must be supplied. ORNL has other facilities and infrastructure to support nuclear fuel development, including post-irradiation examination facilities, analytical chemistry facilities, basic transuranic research laboratories, waste management and treatment facilities, and hot cells and laboratories, including alpha laboratories, of the Radiochemical Engineering Development Center. All of these facilities are now being operated. Most of the support infrastructure and equipment is already in place to accommodate the work. It will be necessary to upgrade and refurbish some of the support equipment and facilities.

The primary buildings and facilities are illustrated in the following descriptions and photographs.

7.2.1 Uranium Fabrication and Characterization

These buildings are suitable for processing and characterization of fuels of various types prior to irradiation.

Building 4508 was originally designed for research and development of processing of uranium and thorium, and a part of the facility was used for the fabrication of plutonium-containing fuels. Although most of the ceramic fuels processing equipment was removed in the 1980s, the building is still suitable for most of the operations to be installed, and some of the building can be used for uranium work. A picture of Building 4508 is shown in Fig. 7-1. Most of which is suitable to conversion to fuels programs since the building was designed with ventilation and contamination zoning for this purpose. It is unlikely that Building 4508 would be used for transuranic fuels processing, fabrication and research.

Figure 7-1. Metals and Ceramics Laboratory Building 4508, ORNL
(Building 4508 is the large building in the right foreground.)

The 7600 Complex originally designed to house the Experimental Gas-Cooled Reactor and shown in Fig. 7-2, was converted in the 1970s for use in processing uranium, particularly for work on reprocessing technology. A small amount of chemical separation work is currently being done in laboratories in the building. The main areas of 7600 Complex could be converted back to processing uranium, since it is equipped with the necessary containment and ventilation. Part of the building is now used for robotics research and a mockup area testing of equipment for the Spallation Neutron Source, an activity scheduled to end in 2004. Part of the high bay area is shown in Fig. 7-3. The building also has two laboratories restricted for use with radioactive materials.

Figure 7-2. 7600 Complex, ORNL
(The offices, high bay area and laboratories are in the center.)

Figure 7-3. High Bay, 7600 Complex, ORNL
(The offices, high bay area and laboratories are in the background)

Building 4501, shown partially in the left foreground of Fig. 7-1, was designed for and can be used for uranium-based processing and research. Figure 7-4 is a photograph of a laboratory in Building 4501.

Figure 7-4. Process Laboratories, Building 4501, ORNL

In addition, other locations can be utilized for uranium fuels work. For example, In addition, some space in Building 3525 could be converted to such use, without impairing the use of the hot cells for post-irradiation examination.

7.2.2 Facilities for DF Fabrication and Characterization

Several buildings have alpha-laboratories or space in which alpha-laboratories can be installed, thus providing a location for preparation, fabrication, and characterization of DF, which contains plutonium and neptunium. Two of these buildings (Buildings 7920 and 7930) are a part of the Radiochemical Engineering Development Center (REDC).

Laboratory space in Building 7920, the primary transuranic processing facility at ORNL, is suitable for the preparation of DF particles. See operations depicted in Fig. 7-5. In this figure, the gloves boxes on the left are being installed for the purpose of preparing particles with Pu, Np, Cm, and Am. Other alpha laboratory space in Building 7930 could be used for fabrication and characterization of DF.

Figure 7-5. Glovebox Labs in Building 7920, ORNL

In addition, other alpha laboratories are suitable for use in characterization and research with small quantities of plutonium and other transuranic isotopes, notably Building 5505, the Transuranium Research Laboratory (TRL), where work is now done on the chemistry and solid state physics of transuranic elements and compounds, see Fig. 7-6. The TRL is well equipped for supporting research to understand the systems and processes for these elements. Capability and expertise of TRL include specialized materials synthesis, high-pressure and high-temperature X-ray diffraction, high-temperature vaporization instrumentation, optical and Raman spectroscopy, high-temperature differential scanning calorimetry, and time-of-flight and trapped ion mass spectrometry. We have studied (and retain capabilities to study) the actinide elements Th through Fm, and routinely handle research quantities of the transuranics, Np Pu, Am and Cm.

Figure 7-6. Transuranium Research Laboratory, Building 5505, ORNL

In addition, some space in Building 3525 could be converted to alpha laboratories, without impairing the use of the hot cells for post-irradiation examination.

7.2.3 Facilities for TF Fabrication and Characterization

The premier facility for processing, fabrication, and characterization of TF is Building 7930, which was originally the Thorium Uranium Recycle Facility, designed for refabrication of nuclear fuels containing high alpha and gamma radiation. See view of the exterior of this building in Fig. 7-7. Thus, with 5.5 feet of concrete shielding and completely remote operation and maintenance systems, TF fuels can be fabricated in Building 7930. See pictures of the interior of the facility in Figs. 7-8 and 7-9. Building 7930 contains very large hot cells designed originally for processing and refabrication of fuels remotely. The main complex of 4 processing cells is approximately 120 feet long and 30 feet wide. The cells are shielded with 5 feet of concrete, and the large central cells are lined with stainless steel. The facility is partially utilized in the transuranium processing program at this time.

In addition to the use of Building 7930, characterization and materials research on TF would be assisted by the use of the materials testing and examination capability of Building 3525, since the fuels of interest must have alpha containment as well as shielding from penetrating gamma radiation. Metallography, electron microscopy, physical measurements, and testing of TF can be done in Building 3525, the facility for post-irradiation examination of fuel.

Figure 7-7. Building 7930, Nuclear Materials Processing Building , ORNL

Figure 7-8. Building 7930, Operating Area-East Side, ORNL

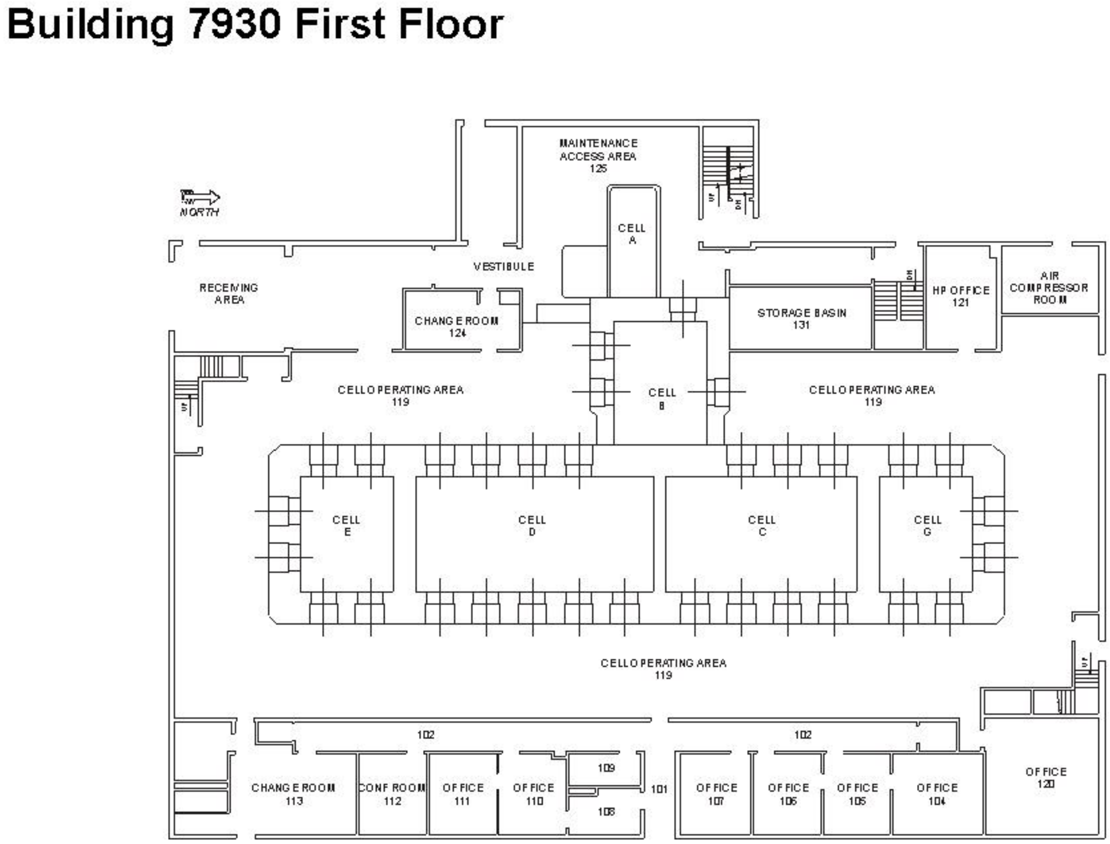

Figure 7-9. Layout of Building 7930, ORNL

7.2.4 Facilities for Irradiation Testing of Fuel

The High Flux Isotope Reactor has been used in the past for irradiation testing of fuel, particularly in support of coated particle fuel development. The complex of consisting of HFIR and the REDC is depicted in Fig. 7-10. A view of the reactor and its storage basin is shown in Fig. 7-11. In addition, the Advanced Test Reactor in Idaho can be used for irradiation tests. Both of these reactors are in operation and are suitable for irradiation tests required by this program plan.

7.2.5 Facilities for Post Irradiation Examination of Fuel

The primary facility for examination of fuels is Building 3525. This facility, which has heavily shielded alpha- containing cells, is equipped with most of the systems required to dismantle and examine fuels. A trained staff for operating this facility is in place. Some upgrades of research equipment and waste handling will be required. See Figs. 7-12 and 7-13.

Figure 7-10. View of HFIR (7910) and REDC Buildings, ORNL
(HFIR is in foreground and Buildings 7920 and 7930 are the large buildings in the background)

Figure 7-11. View of HFIR Reactor viewed through pool, ORNL

Figure 7-12. Building 3525, Metals and Ceramics Division, ORNL

Figure 7-13. Building 3525, Metals and Ceramics Division, ORNL
(View of part of one of three operating areas of the hot cells in Building 3525)

7.3 Facilities Required for Fuel Fabrication Process Development

The fuel fabrication process development will be carried out in several different types of laboratories and facilities as follows:

1. Uranium Fuel Fabrication Facilities-Small Scale
2. DF Fuel Fabrication Facilities-Small Scale (Glove Box)
3. TF Fuel Fabrication Facilities-Small Scale (Hot Cell)

In addition, some operations using only uranium will be pursued to support development of processes for larger scale industrial operations, to develop process changes that are more amenable to remote operation, and to support cold-testing of full-scale equipment necessary for proof-testing (e.g., coating).

Table 7-1 summarizes the principal operations and services that must be provided for recovery of technology and development of fabrication processes for HTGR fuel containing various isotope and for fabrication of irradiation capsules.

Table 7-1. Facilities Required For Fuel Fabrication Process Development

Fuel Type	**Materials**	**Type Facility**	**Primary Operations**	**Technical Services**	**Support Services**
Laboratory or Small Scale Operations					
Uranium	Depleted to Enriched Uranium	Hoods and Controlled Ventilation	Preparation, testing, and assembly of fuel and capsules	Analytical chemistry and metallurgical services	Waste management operations
DF	Plutonium, Neptunium	Glove boxes	Preparation, testing, and assembly of fuel and capsules	Analytical chemistry and metallurgical services	Waste management operations
TF	Pu, Am, Cm, Np	Hot Cells-alpha containment	Preparation, testing, and assembly of fuel and capsules	Analytical chemistry and metallurgical services	Waste management operations
Process Support Operations					
Uranium	Depleted to Enriched U	Hoods and Controlled Ventilation	Selected scaled-up processes	Analytical chemistry and metallurgical services	Waste management operations

7.3.1 Laboratory (Small-Scale) Fabrication Facilities

Small scale facilities and equipment will be required for the three types of fuel. The sections that follow address the requirements for uranium fuel, DF, and TF.

7.3.1.1 Uranium Fuel Fabrication - Small Scale

Table 7-2 lists the process equipment groups and services that must be provided for fabricating fuel and fuel capsules containing only uranium isotopes. The operations to be conducted in the process line would be small-scale operation, involving batches of 200 grams or less of the fuel isotopes. The status of such facilities and equipment at ORNL is shown as existing (E), under development (D), and to be provided (T). Funding for the T items is to be provided by this program.

Table 7-2. Facilities Required for Uranium Fuel Fabrication Process Development

Fuel Isotopes	Component Handled	Type Facility	Primary Operations	Technical Services	Support Services
U-235 and U-238 (Depleted and Enriched combinations)	Uranium containing solutions and bare U compounds	Hoods and controlled ventilation (E)	Sol-gel operations and resin loading (E)	Analytical chemistry and metallurgical services for U-bearing materials (E)	Waste management operations (E)
	Bare U-particles and coated U-particles	Hoods and controlled ventilation (D)	Sintering & Coating operations (D)	Analytical chemistry and metallurgical services (E)	Waste management operations (E)
	Coated particles to compacts with particles	Hoods and controlled ventilation (T)	Fuel Compacting (T)	Analytical chemistry and metallurgical services (E)	Waste management operations (E)
	Fuel compacts to completed capsules	Controlled enclosures and ventilation (T)	Capsule Assembly (T)	Analytical chemistry and metallurgical services (E)	Waste management operations (E)
	From bare particles to fuel compacts	Controlled enclosures and ventilation (T)	Characterization (D) and Quality control (T)	Analytical chemistry and metallurgical services (E)	Waste management operations (E)
	E- existing at ORNL D- under development at ORNL (being designed and installed) T- to be designed, built, and installed				

Facilities now existing at ORNL (notably Buildings 4501 and 4508) can be used for the development of U fabrication processes. Some additional specialized hoods, ventilation control, and process and quality control equipment will need to be added.

Principal items of equipment required (T category) for the above operations include:

1. Equipment for injection of pitch or resin-based matrix materials for compact preparation.
2. Equipment for treating these compact to temperatures up to 1800°C.
3. Measurements not included with primary processes (QC).

Suitable facilities exist at ORNL carry out the essential technical support functions of materials investigations, including metallography and analytical chemistry. In addition, ORNL has in operation waste management functions to handle all wastes from the uranium fuels development.

7.3.1.2 DF Fabrication – Small Scale

Table 7-3 lists the process equipment groups and services that must be provided for fabricating fuel and fuel capsules containing plutonium and neptunium isotopes (driver fuel) and the status of each at ORNL. The operations to be conducted in the process line would be small-scale operation, involving batches of 200 grams or less of the fuel isotopes. The status of such facilities and equipment at ORNL is shown as existing (E), under development (D), and to be provided (T). Funding for the T items is be provided by this program.

Principal items of equipment required (T category) for the above operations include:

1. Sol-gel and/or resin loading equipment,
2. High temperature furnaces for calcination of microspheres,
3. High-temperature coater for applying pyrolytic carbon, SiC, and ZrC,
4. Equipment for injection of pitch or resin-based matrix materials for compact preparation,
5. Equipment for treating compacts to temperatures up to 1800°C,
6. Glove boxes and tools for assembly of capsules,
7. Glove boxes for measurements not included with primary processes (QC).
8. Glove boxes for metallographic examination, including specimen preparation and microscopes and other examination equipment.

These systems would be installed in existing laboratories in ORNL buildings 7920, 7930, and or 3525, which have space suitable to conversion to alpha laboratories. Facilities exist for handling small quantities of CH-TRU (contact handled transuranic) waste, which will be the classification of the waste from the glove boxes. Also, the analytical chemistry functions can be carried out in existing facilities.

Table 7-3. Facilities Required for Driver Fuel Fabrication Process Development

Fuel Isotopes	Component Handled	Type Facility	Primary Operations	Technical Services	Support Services
Pu and Np	Pu-Np containing solutions and bare Pu-Np solid compounds	Glove Boxes (D)	Sol-gel operations (D) and/or resin loading (T)	Analytical chemistry (E) and metallurgical services for Pu-bearing materials (T)	Waste management operations (E)
	Bare Pu-Np particles and coated Pu-Np particles	Glove Boxes (T)	Sintering & Coating operations (T)	Analytical chemistry (E) and metallurgical services (T)	Waste management operations (E)
	Coated particles to compacts with particles	Glove Boxes (T)	Fuel Compacting (T)	Analytical chemistry (E) and metallurgical services (T)	Waste management operations (E)
	Fuel compacts to completed capsules	Glove Boxes (T)	Capsule fabrication (T)	Analytical chemistry (E) and metallurgical services (T)	Waste management operations (E)
	From bare particles to fuel compacts	Glove Boxes (T)	Quality control (T)	Analytical chemistry (E) and metallurgical services (T)	Waste management operations (E)
	E- existing at ORNL D- under development at ORNL (being designed and installed) T- to be designed, built, and installed				

7.3.1.3 TF Fuel Fabrication - Small Scale (Hot Cell)

Table 7-4 lists the process equipment groups and services that must be provided for fabricating fuel and fuel capsules containing plutonium, americium, and curium isotopes (transmutation fuel) and the status of each at ORNL. The operations to be conducted in the process line would be small-scale operation, involving batches of 200 grams or less of the fuel isotopes. The status of such facilities and equipment at ORNL is shown as existing (E), under development (D), and to be provided (T). Funding for the T items is be provided by this program.

Table 7-4. Facilities Required for Transmutation Fuel Fabrication Process Development

Fuel Isotopes	Component Handled	Type Facility	Primary Operations	Technical Services	Support Services
Pu, Am, and Cm	Pu-Am-Cm containing solutions and bare Pu-Am-Cm solid compounds	Alpha Containment Hot Cells (E)	Sol-gel operations (T) and/or resin loading (E)-small scale	Analytical chemistry (E) and metallurgical services for U-bearing materials (T)	Waste management operations (E)
	Bare Pu-Am-Cm particles and coated Pu-Am-Cm particles	Alpha Containment Hot Cells (E)	Sintering and coating operations (T)	Analytical chemistry (E) and metallurgical services (T)	Waste management operations (E)
	Coated Pu-Am-Cm particles to compacts with particles	Alpha Containment Hot Cells (E)	Fuel compacting (T)	Analytical chemistry (E) and metallurgical services (T)	Waste management operations (E)
	Fuel compacts with Pu-Am-Cm to completed capsules	Alpha Containment Hot Cells (E)	Capsule fabrication (T)	Analytical chemistry (E) and metallurgical services (T)	Waste management operations (E)
	From bare particles to fuel compacts	Alpha Containment Hot Cells (E)	Quality control (T)	Analytical chemistry (E) and metallurgical services (T)	Waste management operations (E)
	E- existing at ORNL D- under development at ORNL (being designed and installed) T- to be designed, built, and installed				

Principal items of equipment required (T category) for the above operations include:

1. Sol-gel and/or resin loading equipment,
2. High temperature furnaces for calcination of microspheres,
3. High-temperature coater for applying pyrolytic carbon, SiC, and ZrC,
4. Equipment for injection of pitch or resin-based matrix materials for compact preparation,
5. Equipment for treating compacts to temperatures up to 1800°C,
6. Work stations for assembly of capsules, and
7. Work stations for measurements not included with primary processes (QC).
8. Work stations for metallographic examination, including specimen preparation and microscopes and other examination equipment.

These systems would be installed in existing hot cells at ORNL that are designed for alpha containment. Buildings that are candidates for utilization include Buildings 7920, 7930, and or 3525, all of which have extensive cells and existing analytical chemistry facilities. Facilities exist for handling RH-TRU (remote-handled transuranic) waste and CH-TRU waste, which will be the classifications of waste from these operations. However, the liquid waste processing facilities for Building 3525 will need to be upgraded to handle the wastes from this fuels development program.

7.3.2 Prototype Refabrication Facilities

Activities to support design of industrial-scale equipment and processes will be done in development laboratories. Development of processes on a small scale that are more suitable to scale-up will be done in existing facilities, and will be based on the containment and hazard requirements. Small-scale process development with uranium can proceed in the laboratories previously identified (Building 4501, 4508, or 7601), while work that requires handling the constituents of DF and TF will require at least shielded glove boxes (Buildings 7920, 7930, 3525) as given previously.

Activities requiring large-scale equipment development, such as cold testing of proof-test coater, will be conducted in the existing facilities with uranium with the appropriate containment. Clean walk-in cells exist for testing large coaters in Building 4505, and in other locations. Other labs, such as those in Building 4508, also have large spaces suitable for testing large coaters with depleted uranium or enriched U under special controls.

The support functions of materials science, analytical chemistry, and waste management would be the same ones used for the development of U fabrication processes at small scale. See Section 7.1.1.1 above.

7.4 Facilities for Fuels and Component Characterization and Testing

These facilities are for tests not related to quality control and not requiring integration with fabrication. Included in the fuel characterization and testing category of facilities are low level radiation fuel testing and characterization, high level radiation fuel characterization and testing, reactor test facilities, and post-irradiation, examination, and testing facilities.

7.4.1 Low Radiation Level Laboratories

These facilities are for characterization and testing of uranium bearing fuels and fuel components or with components that do not contain fuel, such as preparation and test of components for capsules. The requirements, facilities, and equipment for U-bearing fuels are summarized in Table 7-5. The status of such facilities and equipment at ORNL is shown as existing (E), under development (D), and to be provided (T). Funding for the T items is be provided by this program.

Table 7-5. Facilities Required for Low Radiation Level Characterization and Testing

Function	Component Handled	Type Facility	Primary Operations	Technical Services	Support Services
Fuel Characterization	U containing solutions and bare U solid compounds	Included in Section 7.1 under Uranium process requirements			
Uranium Fuel Test Facility	U coated particles and compacts	Hoods and controlled ventilation (T)	Thermo-chemical and thermo-physical testing (T)	Analytical chemistry (E) and metallurgical services	Waste management operations (E)
	U coated particles and compacts	Hoods and controlled ventilation (T)	Groundwater testing (T)	Analytical chemistry (E) and metallurgical services (E)	Waste management operations (E)
	U coated particles and compacts	Hoods and controlled ventilation (T)	Oxidation testing (T)	Analytical chemistry (E) and metallurgical services (E)	Waste management operations (E)
Irradiation Test Component Preparation	No fuel components present	Non-radiation laboratory space (T)	Capsule component assembly and measurement (E)	Analytical chemistry (E) and metallurgical services (E)	Waste management operations (E)
	E- existing at ORNL D- under development at ORNL (being designed and installed) T- to be designed, built, and installed				

7.4.1.1 Fuel Characterization Facilities

All of the fuel characterization facilities and equipment was covered in the fuel fabrication process facilities (Section 7.3) above. Those facilities will meet the requirements for all bare fuel and compact characterization.

7.4.1.2 Low Level Fuel Out-of-Reactor Test Facilities

Out-of-reactor test facilities for tests with uranium bearing fuel and fuel components include equipment for measurement of thermochemical and physical properties of fuel compacts and for oxidation, burning, and water environment. These latter facilities must be set-up for both short-term tests (hours to days) and long-term tests (years)

7.4.1.3 Irradiation Test Preparation Facilities

These facilities would be for preparation and test of capsule hardware prior to introduction of fuel. Existing facilities would be utilized. No significant capital expenditures are required. The work on capsules for assembly with uranium fuel and testing thereafter must be done in controlled ventilation, and these facilities are provided Section 7.3.1.1 above.

7.4.2 High Radiation Level Laboratories

These facilities are for working with DF and TF bearing fuels and fuel components. Components that do not contain fuel, such as preparation and test of components for capsules, will be prepared in the low-level facilities discussed above. The requirements, facilities, and equipment for DF and TF are summarized in Table 7-6. The status of such facilities and equipment at ORNL is shown as existing (E), under development (D), and to be provided (T). Funding for the T items is be provided by this program.

7.4.2.1 Fuel Characterization Facilities- DF and TF

All of the fuel characterization facilities and equipment was covered in the fuel fabrication process facilities (Section 7.3) above. Those facilities will meet the requirements for all DF and TF bare fuel and compact characterization.

7.4.2.2 Fuel Out-of-Reactor Test Facilities- DF and TF

Out-of-reactor test facilities for tests with DF and TF and transuranic fuel components include equipment for measurement of thermochemical and physical properties of fuel compacts and for oxidation, burning, and water environment. Facilities with glove box containment and hot cell containment for DF and TF, respectively, must be set-up for both short-term tests (hours to days) and long-term tests (years).

7.4.2.3 Fuel Irradiation Test Preparation Facilities- DF and TF

Preparation of capsule components will be done in existing non-radiation laboratories and the same facility for all irradiation capsules.

7.4.3 Reactor Test Facilities

Reactor test facilities envisioned for use at the present time are the High Flux Isotope Reactor (HFIR) and the Advanced Test Reactor (ATR) which have been previously used for testing HTGR fuels. Proven, existing single-cell instrumented irradiation designs and capabilities exist at HFIR and ATR. A new multi-cell capsule is being designed for the ATR which is planned to be used in the irradiation tests described in Section 6.

No modification or expenditure is required at either HFIR or ATR.

Table 7-6. Facilities Required for DF and TF Characterization and Testing

Fuel Isotopes	**Component Handled**	**Type Facility**	**Primary Tests**	**Technical Services**	**Support Services**
Fuel Characterization	Pu-Am-Cm containing solutions and bare Pu-Am-Cm solid compounds	Requirements contained in Section 7.1 for DF and TF processing.			
DF Test Facility	Coated Pu-Np particles and compacts	Glove Boxes (T)	Thermo-chemical and thermo-physical testing (T)	Analytical chemistry (E) and metallurgical services	Waste management operations (E)
	Coated Pu-Np particles and compacts	Glove Boxes (T)	Groundwater testing (T)	Analytical chemistry (E) and metallurgical services	Waste management operations (E)
	Coated Pu-Np particles and compacts	Glove Boxes (T)	Long term air stability testing (T)	Analytical chemistry (E) and metallurgical services	Waste management operations (E)
TF Test Facility	Coated Pu-Am-Cm particles and compacts	Alpha Containment Hot Cells (E)	Thermo-chemical and thermo-physical testing (T)	Analytical chemistry (E) and metallurgical services	Waste management operations (E)
	Coated Pu-Am-Cm particles and compacts	Alpha Containment Hot Cells (E)	Groundwater testing (T)	Analytical chemistry (E) and metallurgical services	Waste management operations (E)
	Coated Pu-Am-Cm particles and compacts	Alpha Containment Hot Cells (E)	Long term air stability testing (T)	Analytical chemistry (E) and metallurgical services	Waste management operations (E)
Irradiation Test Component Preparation	No fuel components present	Same facilities as included under Section 7.2.1 above. See Table 5.			
	E- existing at ORNL D- under development at ORNL (being designed and installed) T- to be designed, built, and installed				

7.4.4 Post Irradiation Examination and Test Facilities

Post irradiation examination and test facilities required are summarized along with the status of such facilities in Table 7-7. The status of such facilities and equipment at ORNL is shown as existing (E), under development (D), and to be provided (T). Funding for the T items is be provided by this program.

Table 7-7. Post Irradiation Examination and Test Facilities

Fuel Isotopes	Component Handled	Type Facility	Primary Exams And Tests	Technical Services	Support Services
Post Irradiation Examination of U, DF, and TF	All irradiated fuel and graphite components	Alpha Containment Hot Cells (E)	Disassembly, Materials examination and microscopy (E)	Analytical chemistry (E) and metallurgical services	Waste management operations (E)
DF Test Facility	All irradiated fuel and graphite components	Alpha Containment Hot Cells (E)	Themo-chemical and thermo-physical testing (T)	Analytical chemistry (E) and metallurgical services	Waste management operations (E)
	All irradiated fuel and graphite components	Alpha Containment Hot Cells (E)	Groundwater testing (T)	Analytical chemistry (E) and metallurgical services	Waste management operations (E)
	All irradiated fuel and graphite components	Alpha Containment Hot Cells (E)	Long term air stability testing (T)	Analytical chemistry (E) and metallurgical services	Waste management operations (E)
Accident Test Facility	All irradiated fuel and graphite components	Alpha Containment Hot Cells (E)	Themo-chemical and thermo-physical testing (T)	Analytical chemistry (E) and metallurgical services	Waste management operations (E)
	E- existing at ORNL D- under development at ORNL (being designed and installed) T- to be designed, built, and installed				

7.4.4.1 ORNL Post-Irradiation Examination Laboratory

The ORNL Post-Irradiation Examination Laboratory (Building 3525) is presently equipped to carry out the various functions associated with postirradiation capsule disassembly and the subsequent examination of capsule components, fuel compacts, and fuel particles. These operations include disassembly, sectioning, radiography, metallography, dimensional measurements, and waste handling. The only deficiency is the radioactive liquid waste handling provisions, which will need to be improved, through installation of a collection tank in one of the cells. A double-walled line with a pump must be added from the tank to a location near the building, where it would connect to the ORNL liquid waste system.

7.4.4.2 Post-Irradiation Ground Water Test Facility

Equipment in a hot cell in either Building 7930 or 3525 will be needed to carry out long-term test of irradiated fuel in ground water to simulate repository conditions. Because of the location of examination equipment there, Building 3025 would be preferred.

7.4.4.3 Post-Irradiation Accident Test Facility

Equipment in a hot cell in Building 7930 or 3525 will be required to carry out heating tests that will simulate accident conditions.

8. Quality Assurance Program

The activities described in the FDP shall be performed in compliance with the Quality Assurance Program Plan, APT-PPO-0002 – Revision 0, which was issued for the Accelerator Production of Tritium Project. This plan uses the management criteria contained in 10CFR830.120, “Quality Assurance Requirements,” and DOE Order 5700.6C, “Quality Assurance.”

Activities and associated equipment (A&AE) for fuel development are classified as having the potential for nuclear hazards or not. Thus the A&AE for this task are grouped into four classifications:

Safety-class: those A&AE that accident analysis indicates are needed to prevent accident consequences from exceeding Safety Analysis Report evaluation guidelines. Safety-class designation has been traditionally reserved for A&AE needed for public protection. This designation carries with it the most stringent requirements.

Safety-significant: those A&AE of particular importance to defense-in-depth or worker safety as determined by hazard analysis. Control of safety-significant A&AE does not require meeting the level of stringency associated with safety-class A&AE.

Production support: those A&AE not classed as safety-class or safety-significant but determined to be necessary to support the fuel development task. The rigor of application of QA activities and functions for these A&AE is dependent on such factors as investment, availability of replacement parts, length of replacement time, consequences of failure.

General services: those A&AE not classed as safety-class, safety-significant, or production support. The rigor of application of QA activities and functions for these A&AE shall be determined on a case-by-case basis.

Quality activities in general shall implement the requirements of ANSI/ISO/ASQC Q9001-1994, “Quality Systems – Model for Quality Assurance in Design, Development, Production, Installation, and Servicing,” as appropriate for fuel development activities and associated equipment.

In addition, quality activities involving A&AE classified as safety-class and safety-significant shall implement the requirements of ASME NQA-1-1994, “Quality Assurance Requirements for Nuclear Facilities Applications,” as appropriate to the activity.

9. Schedule and Cost for the Program

The cost and schedule for the program are arranged in a work breakdown structure (WBS) for the major elements of the program, and are traceable to the program task descriptions and requirements through the WBS numbers. The estimated costs are given for each fiscal year. Discussion of dates in this section is in terms of calendar year.

9.1 Summary of the Schedule and Cost Estimate

The schedule for the program is driven by the supply of technology and materials for DF and TF. The schedule is based on the sequencing of events that are feasible by building on technology blocks (a technology driven schedule) with personnel that can be made available (constrained by personnel availability). The program is integrated into the overall logic of the demonstration reactor. This overall schedule is shown in Fig. 9-1.

The schedule is aggressive and will depend on early funding of equipment and its installation. As will be seen, it is possible to staff the program initially with a substantial number of personnel who have done identical or very similar tasks in the past. Their knowledge will be transferred to less experienced personnel as the program progresses.

The schedule shows the Demonstration Modular Helium Reactor construction completion by the end of 2010. The fuel demonstration for DF and TF, giving confidence that both will perform satisfactorily under nominal reactor service conditions, is scheduled for completion by mid-2008. Proof-of-principle of DF is scheduled for completion in 2011; and for TF, in 2012. Validation tests of the DF and TF will be done by mid-2015. This sequencing of data gathering and proof testing will allow the fabrication of and irradiation of DF in the demonstration reactor starting in 2013, with TF starting in 2014. A modular fabrication plant for DF-TF fuel will have been designed and built in the 2010-2016 time frame. A partial core loading, based on the lead tests of these assemblies would be prepared in the 2016-2018 time frame, with irradiation starting in 2019.

The total cost of the program through validation of the developed fuel is estimated to be $216 Million. This does not include the commercial phases of the program- that is, the cost of fabricating test elements for the demonstration reactor or for the modular commercial fuel fabrication facility required for fabrication of a partial or full core of demonstration reactor fuel elements.

The cost of the program through demonstration of the feasibility of both DF and TF is estimated at $135 Million. This part of the program will be complete by mid-2008. The cost trend by year is depicted graphically in Fig. 9-2.

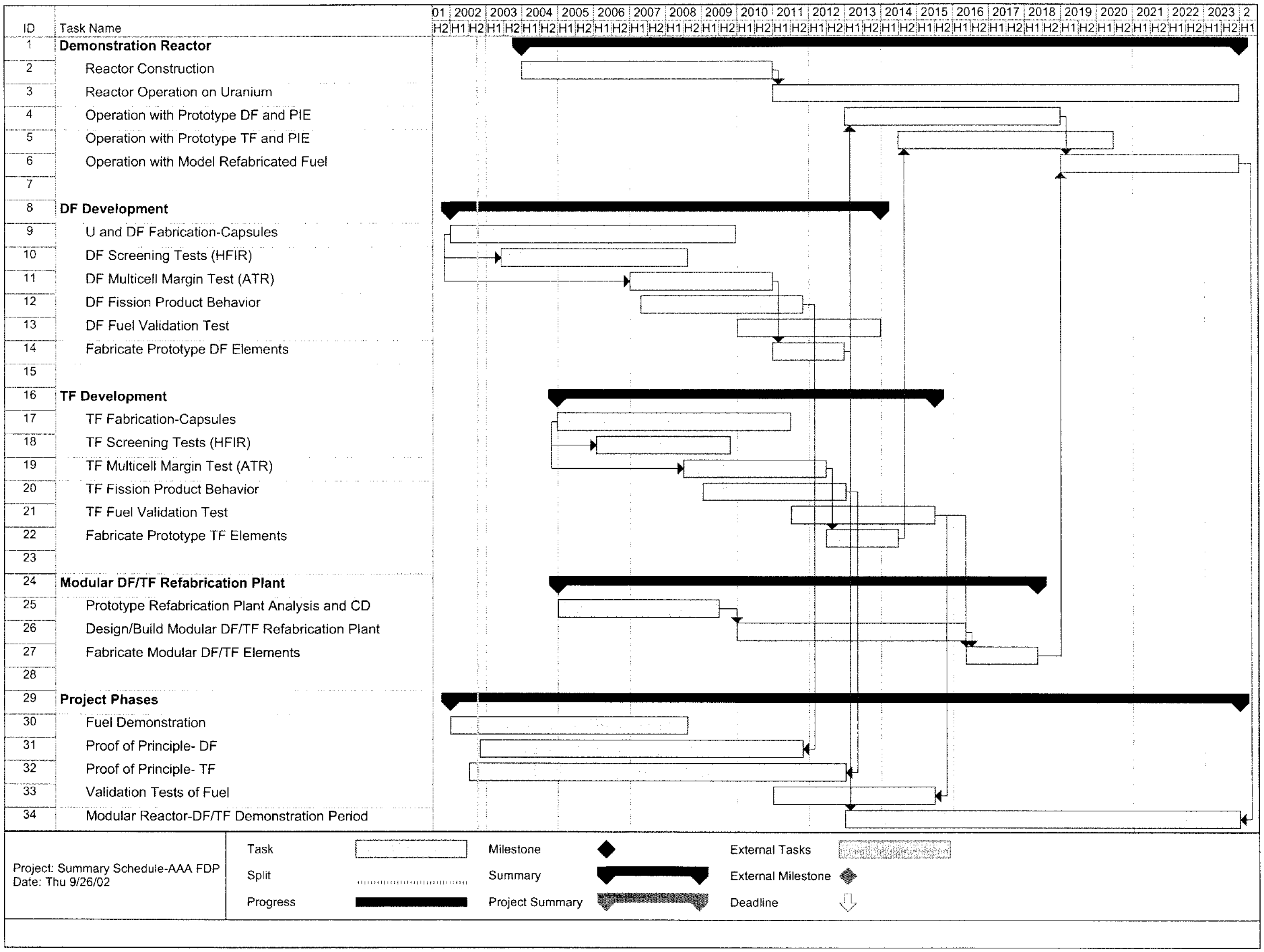

Figure 9-1. Summary Schedule

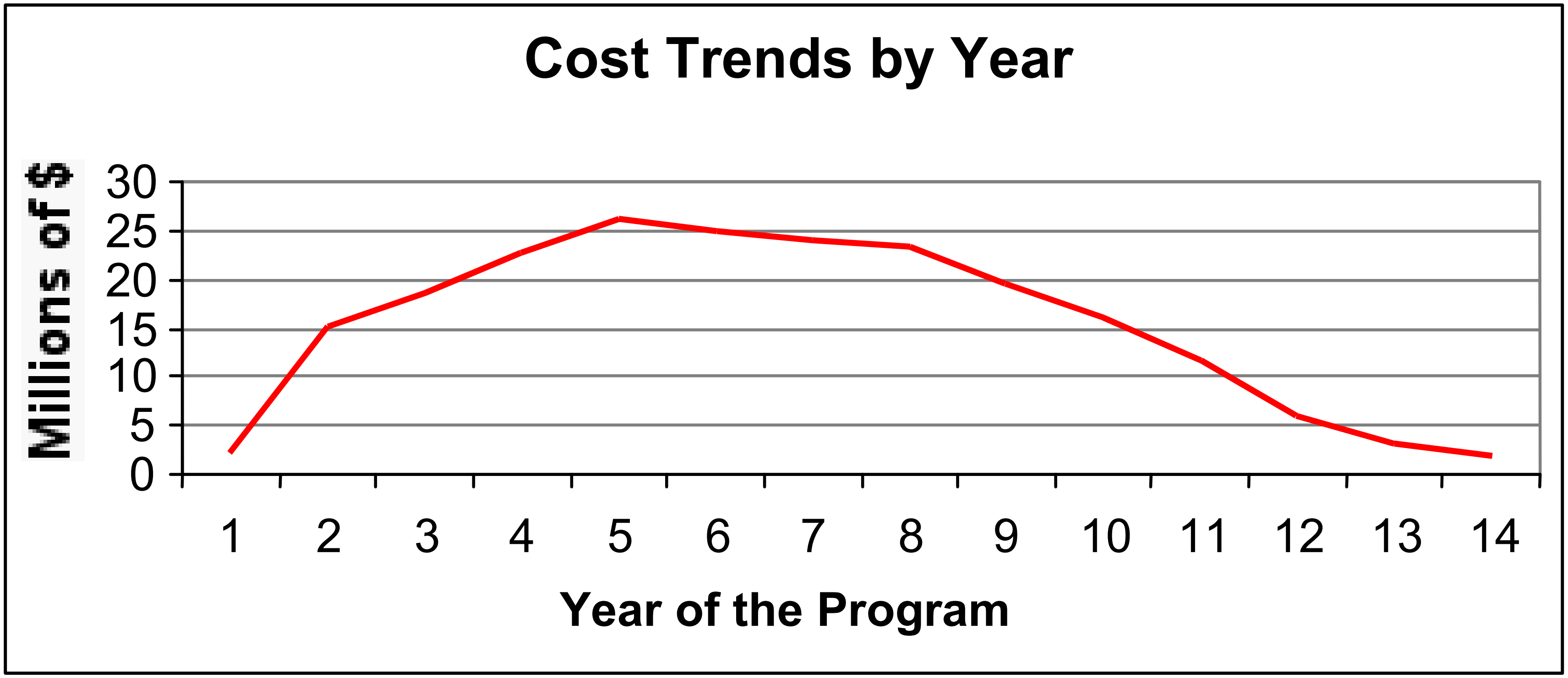

Figure 9-2. Cost Trends by Year

9.2 Detailed Schedule

The detailed schedule is given in Appendix C. The schedule indicates the logic of the program, starting with the development of facilities for the fabrication of uranium fuel, and then proceeding to the development of the fuel processes for DF and TF alternatives. After development of the processes, fuel is fabricated for testing, capsules are built and irradiated, and the fuel is examined and tested for accident conditions by post irradiation heating tests. Other activities involve determination of the properties and behavior of the fuel out-of-reactor, fission product transport, and behavior of spent fuel under long-term storage conditions. The complete capability for fabrication of the fuels addressed by this program will be available by the end of 2004, thus delivering all of the capability in a period of approximately 3 years. (This is accomplished by fabricating small quantities of TF and putting them in compacts with other fuel, such as DF or U.) This strategy allows the program to deliver all types of fuel for the various capsules by early 2006. The capability for preparing larger quantities of TF will be available by mid-2009, in time to produce fuel for the TF validation capsule.

9.2.1 Fuel Fabrication Process Development (WBS 1.1)

This work involves the establishment of facilities for the processing of the various kinds of fuel. After establishment of each facility, the staff will develop the processes and prepare fuel for irradiation and other research to determine the properties of the materials.

9.2.1.1 Uranium Process Line (WBS 1.1.1)

The capability for making uranium kernels and for coating kernels is being established in 2002. Compacting equipment and quality control equipment will be in place by the end of 2003. Thus, operation with both depleted and enriched uranium should be possible by the end of 2003, making possible delivery of coated particles for irradiation in late 2003, compacts in early 2004. Preparation of alternative coatings (ZrC) will also be scheduled in this time frame. In addition to the work for recovery of technology developed in the past, work for development of equipment for remote fabrication at large (prototypic) scale will be carried out in the 2004-2008 time frame. The work will be done with uranium with equipment designed for remote operation.

9.2.1.2 Driver Fuel Process Line (WBS 1.1.2)

The capability, to manufacture Driver Fuel will be achieved by installing equipment in an existing laboratory, will utilize glove boxes for all operations. Design, procurement, and installation of the processes are scheduled to start in 2003 and will be complete by mid-2004. Thus, production of kernels containing Pu and Np is scheduled to start in late 2003, with production of fuel for testing commencing in 2004.

9.2.1.3 Transmutation Fuel Line (WBS 1.1.3)

The capability, to manufacture Transmutation Fuel will be achieved by installing equipment in an existing hot cell facility, will require shielding for protection from gamma and alpha radiation. The strategy, as mentioned previously, is to start with the preparation of TF kernels in a remotely operated facility and to coat small quantities of them in the DF line, with subsequent processing in the DF line in combination with other particles. Thus, the TF kernel preparation line is required early and will be designed, installed, and ready for operation by the end of 2004. The design and installation of equipment to perform the fabrication operation for larger quantities of TF requiring remote operation will be done by mid 2009. Thus, production of kernels containing larger quantities of Am-Cm is scheduled to start in 2009. Prior to the production of fuel with full loading of Am and Cm, starting in 2009, sparse fuel (containing very small quantities of these isotopes) will have been produced in the driver fuel line in the 2005-2008 time frame.

9.2.1.4 Prototypic Scale Refabrication Analysis (WBS 1.1.4)

This work is scheduled to start in 2005, following experience in setting up and operating laboratory-scale systems for DF and TF, which require, respectively, glove box and hot cell environments. Following the completion of preliminary conceptual design of an industrial scale fuel fabrication plant for DF and TF, starting in 2008 and ending in 2009, the program staff will be able to provide more accurate economic and cost analyses for program decisions.

9.2.2 Fuel Materials Development (WBS 1.2)

The fuels material development includes all of the work for characterization of the behavior of fuel and fuel element components under both non-irradiated and irradiated conditions. The irradiation characterization schedule starts with the irradiation of uranium-based fuel. As the driver and transmutation fuel becomes available, these are added to the irradiation program. Each irradiation includes sequential activities of capsule safety analysis, fabrication, irradiation, and post irradiation examination. Post-irradiation heating and examination are also included. All of this work is included under WBS 1.2.1 for driver fuel evaluation and testing and WBS 1.2.2 for transmutation fuel evaluation and testing. The work on uranium fuels is included in WBS 1.2.1.

9.2.2.1 Driver Fuel Evaluation and Testing (WBS 1.2.1)

The driver fuel evaluation and testing includes both activities for non-irradiated and irradiated properties of fuels containing Pu and Np, as well as some work on recovery of information on uranium-base fuel.

9.2.2.1.1 Non-irradiation Evaluation of Driver Fuel (WBS 1.2.1.1)

All of the fuels are characterized for their thermochemical and thermophysical behavior, both theoretically and experimentally. The activity starts in 2002 with preliminary theoretical consideration of the behavior of Pu-NP systems (as compared to uranium systems), to predict behavior during processing and irradiation. This activity formally merges into combined experimental and theoretical work in 2004, and the work is extended through mid-2009 in two activities involving particles and compacts. This work is scheduled to supply technology and data for interpretation of irradiation results and behavior.

9.2.2.1.2 Driver Fuel Irradiation Testing (WBS 1.2.1.2)

This work area of the program starts with the irradiation of uranium-based fuel in 2003 with the first capsules. Each of the capsules has activities that last approximately 3 years. This is broken down into approximately 1 year of capsule preparation, 1 year for irradiation including typically 10 months of irradiation, and 1 year for post-irradiation examination and post-irradiation accident condition heating tests. As the driver fuels become available additional capsules are added to determine the properties and statistics of fuel behavior, as well as to test variations in the fuel systems, such as employment of ZrC coatings. The preliminary irradiation on DF start in 2005, and information and data on irradiation behavior will be coming out of the program in 2008. More tests for determining statistical behavior of reference fuel is scheduled for the 2007 to 2013 period.

9.2.2.2 Transmutation Fuel Evaluation and Testing (WBS 1.2.2

The transmutation fuel evaluation and testing work includes both activities for non-irradiated and irradiated properties of fuels containing Am and Cm.

9.2.2.2.1 Non-irradiation Evaluation of Transmutation Fuel (WBS 1.2.2.1)

All of the fuels are characterized for their thermochemical and thermophysical behavior, both theoretically and experimentally. The activity starts in 2002 with preliminary theoretical consideration of the behavior of Am-Cm-containing fuel systems, to predict behavior during processing and irradiation. This activity formally merges into combined experimental and theoretical work in 2006 (when the materials are available), and the work is extended through 2009 in two activities involving particles and compacts. This work is scheduled to supply technology and data for interpretation of irradiation results and behavior.

9.2.2.2.2 Transmutation Fuel Irradiation Testing (WBS 1.2.2.2)

As with the capsules for driver fuel, each of the capsules have activities that last approximately 3 years. This is broken down into approximately 1 year of capsule preparation, 1 year for irradiation including typically 10 months of irradiation, and 1 year for post-irradiation examination and post-irradiation accident condition heating tests. As the transmutation fuels become available additional capsules are added to determine the properties and statistics of fuel behavior. The preliminary irradiation of TF (in the first two capsules) will start in 2006 using fuel spiked with just very small quantities of Am-Cm-containing fuel particles, and information and data on irradiation behavior of these sparse transmutation fuels will be coming out of the program in 2009. Tests of Am-Cm fuel made in hot cells with full population of TF particles are scheduled to start by 2011, with initial data coming from the program in 2015.

9.2.3 Fission Product and Actinide Transport (WBS 1.3)

Fission product and actinide behavior for various fuel element components is scheduled as early as the availability of DF and TF allows. Four capsules are to be started as early as possible with some spread in the schedule to allow some leveling of the effort. The capsules are started in 2007 and results from them are complete by 2013. Each capsule activity involves approximately a four-year period, divided into 1-year preparation, 1.5-year irradiation, and 1.5-year post-irradiation examination (PIE). The fuel from each of the tests will be tested for accident conditions by post-irradiation heating. These heating tests are shown in parallel with the PIE effort because particles for this work will be available at the beginning of the PIE period.

9.2.3.1 Retention in Fuel Compacts (WBS 1.3.1)

These tests (one each for DF and TF) are scheduled to start with capsule fabrication in 2007. Data from these tests, including the accident condition testing, will be coming from the program in 2011 and 2013.

9.2.3.2 Retention in "Fuel Bodies" (WBS 1.3.2)

These tests are scheduled to start with capsule fabrication in 2007. Data from these tests, including the action condition testing, will be coming from the program in 2011 and 2012.

9.2.4 Spent Fuel Disposal Technology (WBS 1.4)

These tests, which will be done on both non-irradiated and irradiated tests of reference fuels, starting soon after the first date of availability of DF (2006) and end with test on irradiated fuel components (TF in 2012). There are three primary elements in the schedule – mechanical integrity of components, fuel component performance in air and/or water, and fission product inventories and migration in fuel components.

9.2.4.1 Mechanical Integrity (WBS 1.4.1)

These tests under simulated repository or storage conditions will start with the various fuel forms in 2006 and be conducted at least through the duration of the program (2012), with periodic removal of thief samples of the leach solutions for radiochemical analysis.

9.2.4.2 Fuel Component Performance (WBS 1.4.2)

Determination of the performance of the fuel components in retaining and fission products under air and water conditions will be done in the program periodically as the components of different batches of materials are provided, commencing in 2006 with non-irradiated materials and ending in 2012 with materials from irradiated TF.

9.2.4.3 Radionuclide Transport in Spent Fuel (WBS 1.4.3)

Determination of the performance of the fuel components in transporting fission products under air and water conditions will be done in the program periodically as the components of different batches of materials are provided, commencing in 2006 with non-irradiated materials and ending in 2012 with materials from irradiated TF.

9.3 Cost Estimate for the Program

9.3.1 Cost Breakdown by Program Tasks

The cost of the program has been estimated from the detailed activities of the schedule, with consideration of the components involved in each activity. The costs are summarized in Table 9-1, and details are given in Appendix D.

9.3.2 Discussion of Estimate

The expenditures are shown by major tasks in Fig. 9-3. The cost is dominated by the irradiation tasks, which account for 47% of the program cost.

The cumulative costs for the program through fuel demonstration is given in Fig. 9-4. The cost of demonstration of DF and TF is $134 Million.

The cumulative costs for the complete program through validation of the fuel in irradiation tests is given in Fig. 9-5.

The breakdown of the total estimate according to type of expense is given in Fig. 9-6, where it strongly depicts that the program has very large direct expense for services, materials, installation labor, and operational costs not related to direct personnel.

9.3.3 Discussion of Staffing

The program staffing requirements are given in Fig. 9-7. The program has a rather steep demand for personnel starting in 2003 at a total of 40 and peaking in 2007 at 68. The requirements steadily decline after 2007 until the end of this program scheduled for 2015. Staffing the steep demand at the beginning will be difficult. Our evaluation of this challenge indicates that the staffing can be achieved through the recruitment of ORNL staff with relevant experience, of consultants with directly applicable experience, and by assignment of GA staff to ORNL. As much as one half (14 in 2004) of the engineering and scientific staff (28 total in 2004) can be obtained in this manner. The requirement of 21 technicians can be met with a combination of experienced ORNL staff (10) and new hires. Sufficient support staff and others paid from direct costs appear to be adequate, and additional staffing can be met. We believe the schedule proposed can be met through the forgoing staffing strategy.

Table 9-1. Summary of Cost Components for Program

WBS	Name of Task or Subtask	Duration	Start Date	Finish Date	Total All Costs, $
1	Total Program	5739 days	January-02	December-23	215,649,983
1.1	Fuel Fabrication Process Development	2426 days	March-02	June-11	75,257,675
1.1.1	Base Fuel Fabrication Technology	1706 days	March-02	September-08	17,879,525
1.1.2.	Driver Fuel (DF)	1967 days	June-02	December-09	23,576,675
1.1.3	Transmutation Fuel (TF)	2088 days	July-03	June-11	26,510,475
1.1.4	Prototypic Scale Refabrication Analyses	1170 days	January-05	June-09	7,291,000
1.2	Fuel Materials Development	3369 days	August-02	June-15	100,708,278
1.2.1	Driver Fuel Evaluation and Testing	2976 days	August-02	December-13	59,065,728
1.2.2	Transmutation Fuel Evaluation and Testing	3326 days	October-02	June-15	41,642,550
1.3	Fission Product and Actinide Transport	1491 days	April-07	January-13	32,876,650
1.3.1	Retention in Fuel Particles	1491 days	April-07	January-13	17,850,075
1.3.2	Retention in Fuel Compacts	1349 days	November-07	January-13	15,026,575
1.4	Spent Fuel Disposal Technology	1691 days	December-05	June-12	6,807,380
1.4.1	Mechanical Integrity	1691 days	December-05	June-12	986,500
1.4.2	Long-term Performance of Fuel Components	1691 days	December-05	June-12	4,476,800
1.4.3	Radionuclide Transport	1691 days	December-05	June-12	1,344,080

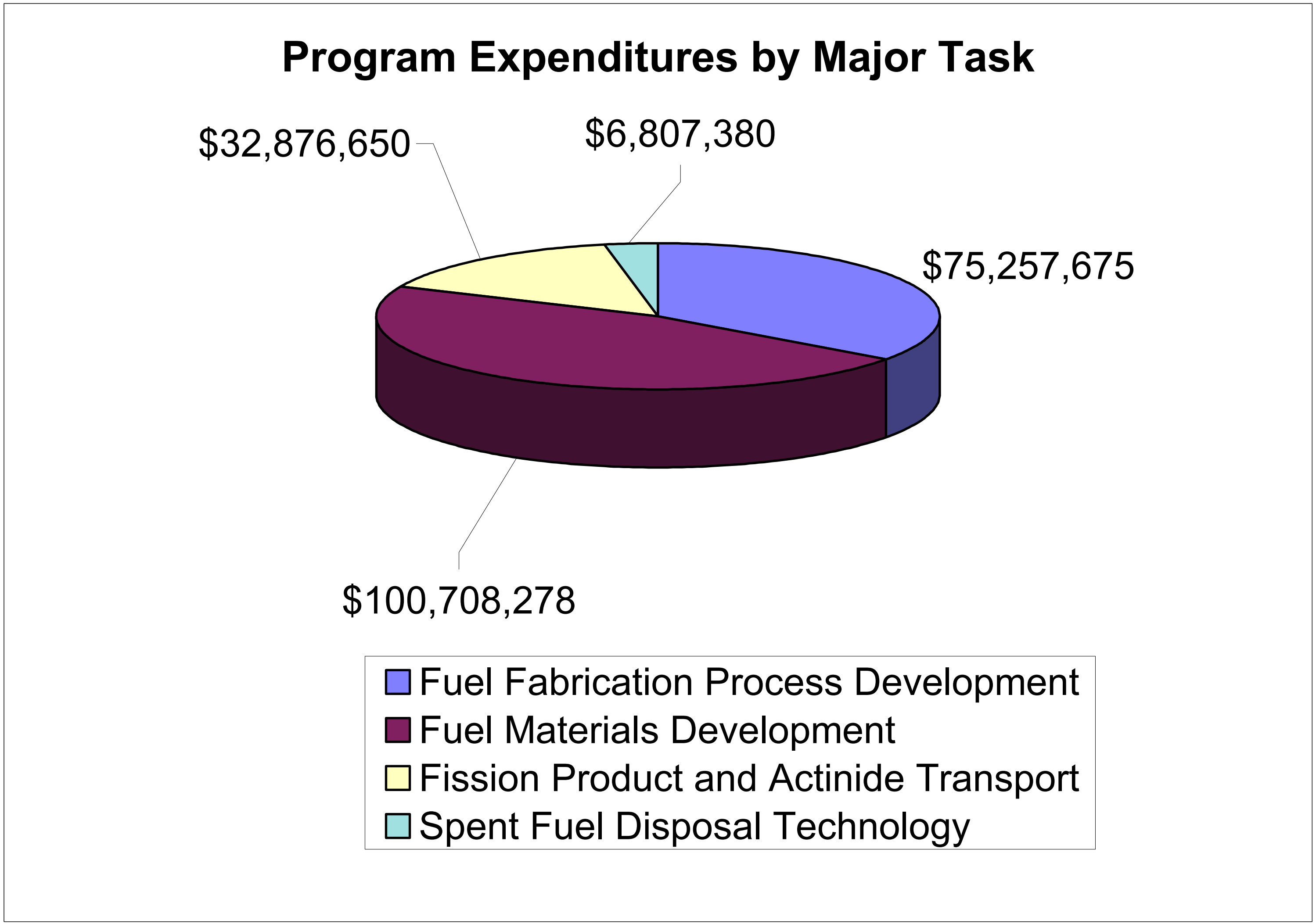

Figure 9-3. Program Expenditures by Major Tasks

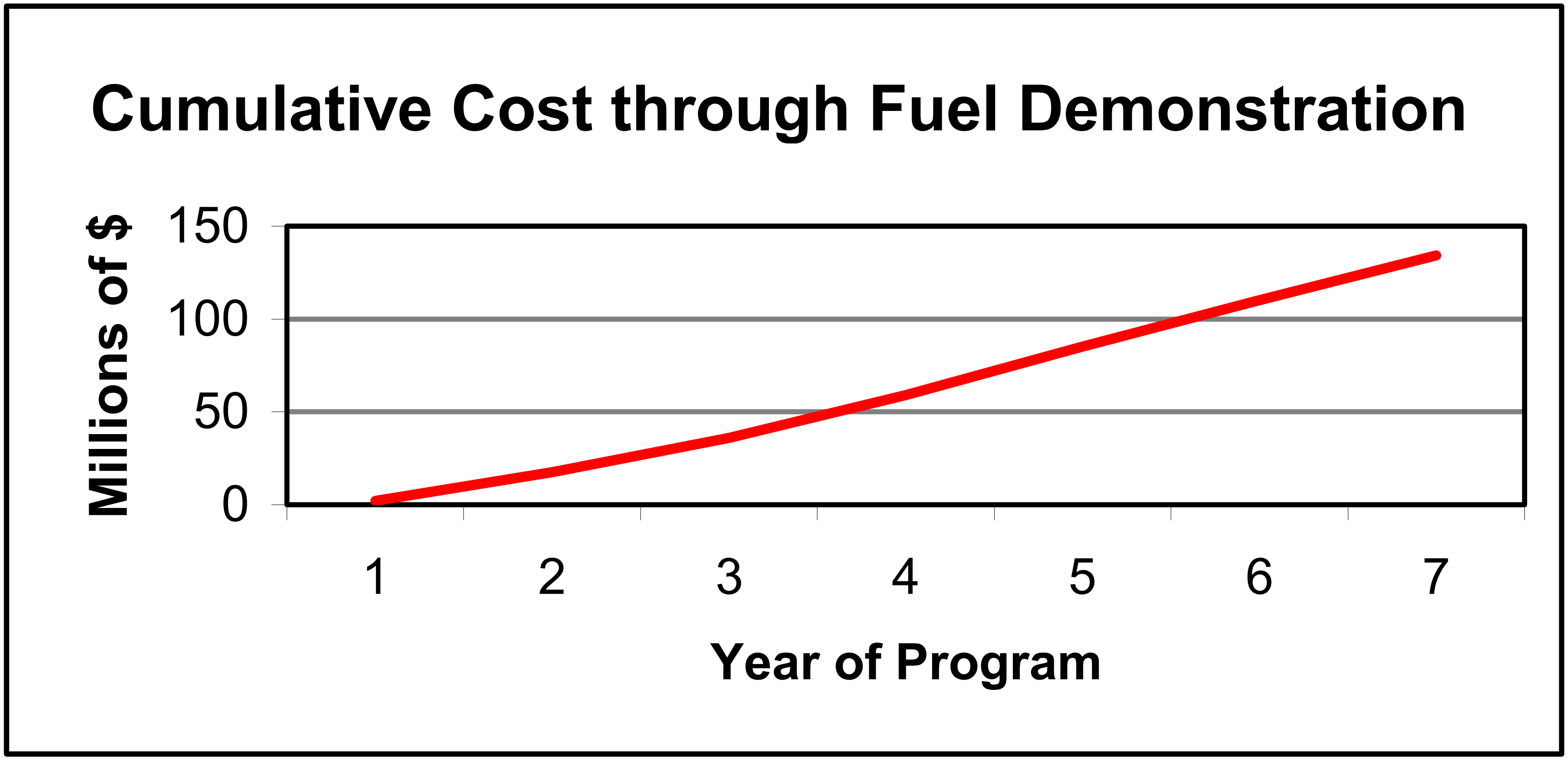

Figure 9-4. Cumulative Costs of Program Through Fuel Demonstration

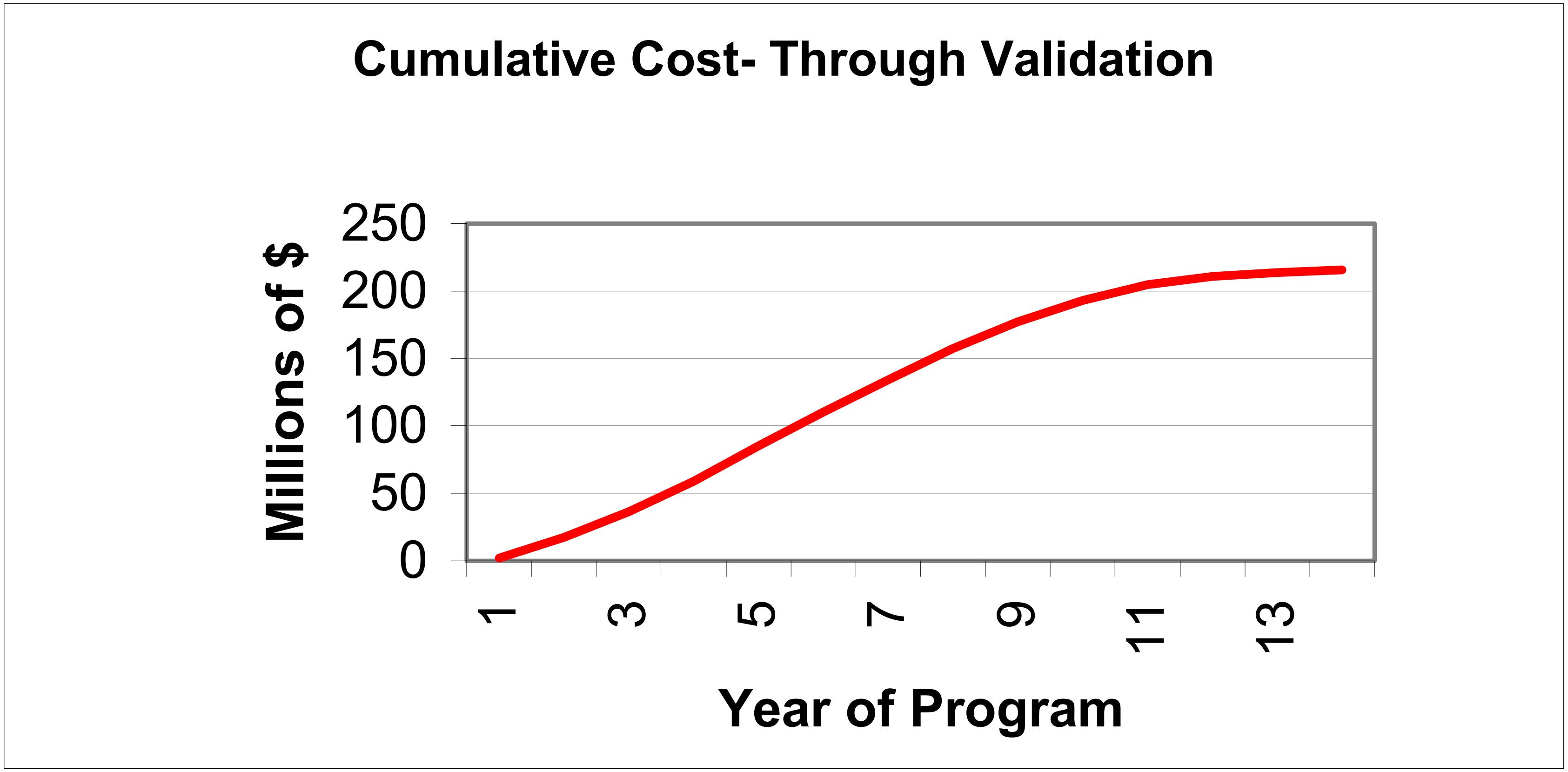

Figure 9-5. Cumulative Cost of Program Through Validation

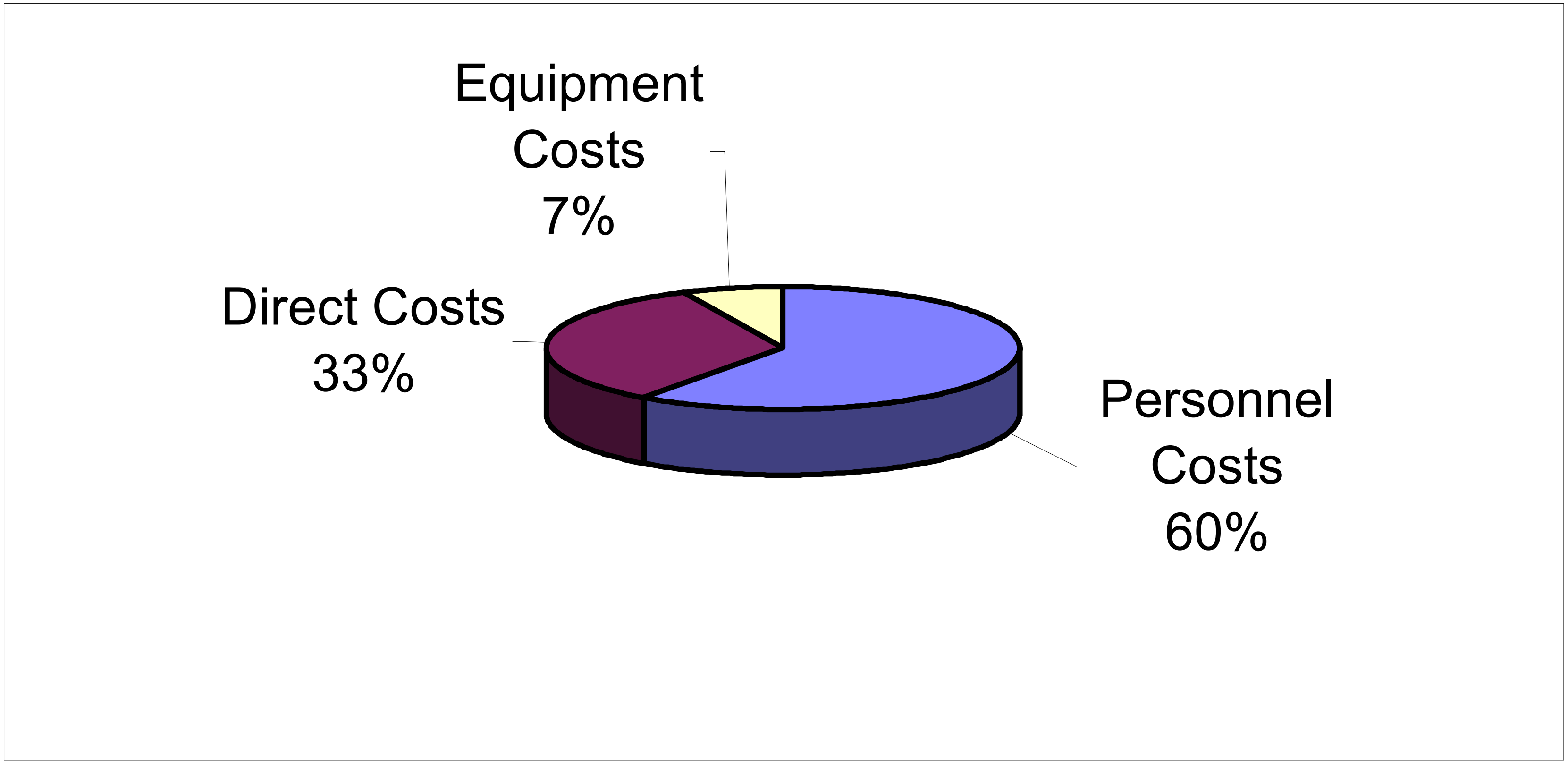

Figure 9-6. Breakdown of Cost Estimate by Type

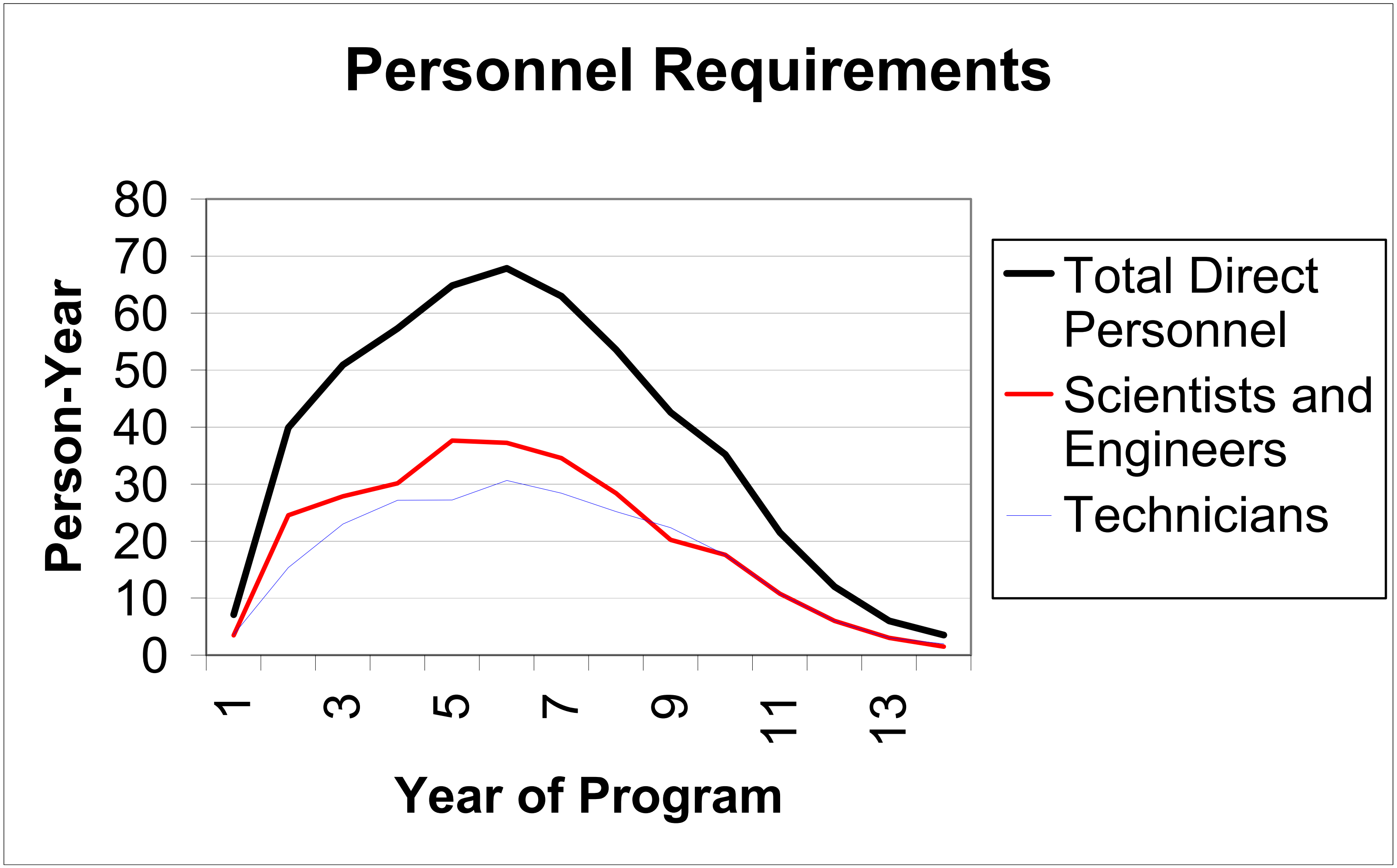

Figure 9-7. Personnel Requirements for the Program

9.3.4 Discussion of Estimate Methodology

The fundamental elements of each activity are direct personnel, direct costs, and capital equipment costs. The total costs (indirect and direct costs) were obtained by using overhead and markup values to account for the normal overheads and management, including cost for operation of office and laboratory space. The determination of each element was as follows:

- Direct engineers and scientists: the number was estimated for each activity and a value per person was used to account for salaries, fringe benefits, overheads, management, and laboratory space. The annual cost for scientists/engineers was estimated as $291,500.

- Technicians: The number was estimated for each activity and a value per person was used to account for salaries, fringe benefits, overheads, management, and laboratory space. The annual cost for scientists/engineers was estimated as $192,500.

- Direct Costs: These are special direct program costs above "normal" costs per person-year. These costs include cost of operation of glove-box facilities and hot cells. Thus, substantial costs are included for the increased costs of operation of such facilities and equipment. Therefore, processing of fuel, some capsule fabrication operations, and post-irradiation work incur large costs for operating in alpha and alpha-gamma containment facilities. (Normal costs include the cost of office rental and "cold" laboratory space.) In addition, the direct costs include irradiation costs for the capsules and special materials for capsules, as well as charges for waste disposal. The cost of installation of equipment of all kinds is included in the direct costs.

- Equipment Costs: These costs are for the purchase of equipment items, which for purposes of this estimate are considered capital items because of their size and expense. Any design and installation costs are included in personnel and direct costs, respectively.

The costs of fuel process development are difficult to determine, as much new technology is to be developed, including the design, procurement, installation, and operation of 3 different fabrication lines, involving hooded, glove-box, and hot cell enclosures. In addition, some scale-up work will be done with uranium. While the uranium work involves recovery of technology that mostly has been done before, and is more confidently estimated, the other activities are on less certain foundation. However, the program plan has enlisted experienced personnel who actually made and irradiated Pu-bearing HTGR fuel in the past, and designed and operated equipment for remote fabrication of HTGR fuel, although the equipment was never operated in a hot cell environment. Thus, their experience has been incorporated as a part of the basis for the process development estimate and establishes a basis for confidence in the estimates for this work.

The costs associated with the irradiation test programs and the post-irradiation work are reasonably well known (WBS 1.2-1.4) because of the wealth of experience, although there are significant extrapolations of past experience to the present time frame, involving inflation and a increased oversight for safety. We believe we have accounted for these effects.

Inherent in this program is the assumption that the objective of proving that these fuels can be attained by success in each of the activities without repeating the activities. Since each activity builds on the other, failure of one can cause delay in the program and additional cost. While there are many ways to recover from failure of some experiments and process attempts, it is nevertheless prudent to consider that this program is optimistic about the degree of success in each step. There is some room to recover within the program cost. Based on the knowledge that we have now, there is every reason to believe that the total cost of this program is approximated closely by our estimate.

10. Deliverables

The program deliverables will be in the form of Letter Reports, Reports, and Fabricated Items. Letter Reports are a less formal communication designed for rapid dissemination of information and task status mainly to program and task workers so that the work direction and near term results can be quickly evaluated and reviewed. They represent the task status at a particular time, are less refined, and may be composed of E-Mail and internal memos.

Reports are formal documentation of the work completed and have an audience beyond that of the immediate project staff and meet an archival need. They provide the long-term documentation of the work, the techniques used in the conduct of the work, and the results of the work.

Fabricated Items are the composite physical components and materials made to satisfy the conduct of the task. In this program they will be mostly irradiation capsules, fuel, and fuel items. They will be discarded after they have served their purpose.

The reports will satisfy the formal program management procedures and QA protocols for the preparation of specific documents to control the planning, execution and evaluation of experimental test programs. Examples of such reports are: test specifications, test plans/procedures, data compilation reports, and test evaluation reports. In simplified (and idealized) terms, the following sequence applies: (1) the cognizant design organization issues a Test Specification; (2) the testing organization prepares Test Plans/Test Procedures that are responsive to the Test Specification; (3) the testing organization performs the subject tests and documents the results in a Data Compilation Report; and (4) the design organization evaluates the test data, including the design implications, and documents the results in a Test Evaluation Report. In reality, the process is iterative, and the roles of the design and testing organizations often overlap significantly (e.g., both the design and testing organizations typically participate in the data evaluation and interpretation).

Because this is an experimental program, the Fabricated Items and physical data are of particular interest. The QA program to support the general needs of the irradiation program and coated particle fuel fabrication is particularly important and should be developed from the onset rather than later in the program to avoid delays and problems. Nuclear hardware is expensive with long lead times and reactor demands are quite rigorous.

Table 10-1 details the deliverables identified to date. They are organized in accordance with the WBS, and the current WBS schedules of Section 9 should be consulted for the expected task completion date.

Table 10-1. Deliverables Identified

WBS	Task Name	Deliverable	Date
1	Fuel Development Schedule	a) Letter Reports on program status b) Quarterly Progress Reports c) Five Year Status Reports d) Final Report e) QA plan for the program f) Cost and Schedule updates g) Planning and Scope Updates	At the completion of program relevant progress, every Quarter, every Five Years, at Project End. The QA plan should be completed before the experimental work begins.
1.1	Fuel Fabrication Process Development	a) Letter Reports on Fuel Fabrication Process b) Quarterly Progress Reports c) Final Report on fuel fabrication methods	At the completion of relevant progress, every Quarter, at task completion.
1.1.1	Basic Fuel Fabrication Technology	a) Letter Report on U Kernel and Coating Work b) TRISO U Particles c) Letter Report on Compacting Work d) U Fuel Compacts e) Report on QC Methods for Coated Particles f) Letter Report on Remote Process Support g) Status of Technology Recovery Report	At the completion of relevant progress, at completion of Task (See current WBS)
1.1.2	Driver Fuel	a) Letter Report on Glovebox Pu/Np Kernel and Coating Work b) TRISO Pu/Np Particles c) Letter Report on Glovebox Compacting Work d) Pu/Np Fuel Compacts	At the completion of relevant progress, at completion of Task (See current WBS)

WBS	Task Name	Deliverable	Date
		e) Report on Feasible QA Methods for Glovebox Coated Particle Fabrication f) Technology and Techniques of Glovebox Coated Particle Fabrication (Technology Description and Summary) Report	
1.1.3	Transmutation Fuel	a) Letter Report on Glovebox Pu/Am/Cm Kernel and Coating Work with Compacting (Spare approach) b) TRISO Pu/Am/Cm Particles (Few, Glovebox) c) Pu/Am/Cm Fuel Compacts (Few, Glovebox) d) Letter Report on Hot Cell Pu/Am/Cm Kernel, Coating, and Compacting Work e) TRISO Pu/Am/Cm Particles (Hot Cell) f) Pu/Am/Cm Fuel Compacts (Hot Cell) g) Report on Feasible QA Methods for Hot Cell Coated Particle Fabrication h) Technology and Techniques of Hot Cell Coated Particle Fabrication Report (Technology Description and Summary)	At the completion of relevant progress, at completion of Task (See current WBS)
1.1.4	Prototypic Scale Refabrication Analyses	a) Letter Reports on Status and Direction of Prototypic Analysis b) Preliminary Design and Economic Analysis of Industrial Production of DF and TF Fuel Report	Quarterly, at completion of Task (See current WBS)
1.2	Fuel Materials Development	a) Letter Reports on Fuel Materials Development b) Quarterly Progress Reports c) Final Report on Fuel Materials Development - Evaluation of Work Accomplished and Implications for Reactor Design	At the completion of relevant progress, every Quarter, at task completion.

WBS	Task Name	Deliverable	Date
1.2.1	Driver Fuel Evaluation and Testing	a) DF Kernel Thermochemical Analysis and Estimation of Performance Report b) Fuel Compact Themophysical Properties Report c) DF Test Specifications d) Single-Cell Irradiation Capsule Design Report (if designed by this program) e) Multi-cell Irradiation Capsule Design Report (if designed by this program) f) Fabricated Irradiation Capsules g) DF Fuel Screening Experimental Plan for all Three Phases h) Results of Phase I Screening – Irradiation and PIE Data Report i) Results of Phase II Screening – Irradiation and PIE Data Report j) Results of Phase III Screening – Irradiation and PIE Data Report k) Report Evaluating SiC and ZrC as Candidate Coating Materials and Validation Status of Selected Option	At the completion of relevant progress, at completion of Task (See current WBS)
1.2.2	Transmutation Fuel Evaluation and Testing	a) TF Kernel Thermochemical Analysis and Estimation of Performance Report b) Fuel Compact Themophysical Properties (if different from 1.2.1) Report c) TF Test Specifications Report d) TF Fuel Screening Experimental Plan for All Three Phases e) Fabricated Irradiation Capsules	At the completion of relevant progress, at completion of Task (See current WBS)

WBS	Task Name	Deliverable	Date
		f) Results of Phase I Screening – Irradiation and PIE Data Report g) Results of Phase II Screening – Irradiation and PIE Data Report h) Results of Phase III Screening – Irradiation and PIE Data Report i) Report Evaluating Candidate Fuel Designs for TF Data Particles and Validation Status of Selected Option	
1.3	Fission Product and Actinide Transport	a) Letter Reports on Fission Product and Actinide Transport b) Quarterly Progress Reports c) Final Report on Fission Product and Actinide Transport – Evaluation of Work Accomplished and Implications for Reactor Design	At the completion of relevant progress, every Quarter, at task completion.
1.3.1 & 1.3.2	Retention in Fuel Particles and Retention in Fuel Compacts (These tasks will run concurrently)	a) Fission Product Chemical Forms and Implications Report b) Fission Product Transport Test Specification Report c) Experimental Plan for Fission Product Transport Report d) Irradiation Capsule Design and Predictions Report e) Fabricated Irradiation Capsules f) DTF & Laser drilled particles g) Compacts with DTF or Laser drilled particles h) Transport Specimens i) Fission Product Transport Under Normal Conditions in Fuel, Graphite, and Core Internals Data Report j) Fission Product Transport Under Accident Conditions in Fuel, Graphite, and Core Internals Data Report	At the completion of relevant progress, at completion of Task (See current WBS)

WBS	Task Name	Deliverable	Date
		k) Comparison Between Models and Experiments – Validation Issues Report	
1.4	Spent Fuel Disposal Technology	a) Letter Reports on Spent Fuel Disposal Technology b) Test specifications c) Test plans/procedures d) Quarterly Progress Reports e) Final Report on Spent Fuel Disposal Technology – Evaluation of Work Accomplished and Implications for Concept	At the completion of relevant progress, at completion of Task (See current WBS)
1.4.1	Long-Term Performance of Fuel Components	a) Test specifications b) Test plans/procedures c) Spent Fuel Performance in Air Report d) Spent Fuel Performance in Groundwater Report	At the completion of relevant progress, at completion of Task (See current WBS)
1.4.2	Radionuclide Transport	a) Test specifications b) Test plans/procedures c) Radionuclide Inventory Report d) Radionuclide Leaching Rate from Fuel Components Report	At the completion of relevant progress, at completion of Task (See current WBS)

Appendix A: Fuel/Fission Product Design Data Needs

As summarized in Section 5, a complete set of fuel/fission product Design Data Needs for the DB-MHR in the prescribed format is included in this appendix.

DATE: 09/27/02

"AAA" Fuel Program
DRIVER FUEL KERNEL PROCESS DEVELOPMENT
DDN RTW.01.01

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The $PuO_{1.7}/NpO_x$ kernels are to be manufactured by an internal gelation process. The existing process steps, as demonstrated on a laboratory scale, must be optimized with respect to flowsheet conditions, scaled up and demonstrated in full size equipment units to be certain that scaling effects are properly treated. The process details, including yields and costs, must be quantified for later transition to a commercial application. The scope of this DDN is to include the operations necessary for transformation of a separations plant product stream into a sintered microsphere suitable for TRISO coating. Solution preparation steps, such as concentration, denitration, and neutralization adjustment, are part of this process.

1.1 Summary of Functions/Assumptions

"Retain Radionuclides in Fuel Particles," Assumptions: Processes are available for manufacturing oxide-based fuel kernels for high integrity coated fuel particles.

1.2 Current Data Base Summary

A traditional sol-gel process (developed at ORNL) was used to manufacture the PuO_{2-x} and $(Th,Pu)O_{2-x}$ kernels that were irradiated in Peach Bottom fuel test element FTE-13 (Ref. 1). The substoichiometric oxygen-to-plutonium ratios were obtained by heat treating in pure hydrogen at 1750 ^oC.

Belgo Nucleaire manufactured 500-µm PuO_x kernels which were diluted with carbon ($C/PuO_2 = \sim 20$) and subsequently TRISO coated for irradiation in the Dragon HTR and Studsvik (e.g., Ref. 2). The kernels were prepared by both sol gel and powder agglomeration processes. Considerable emphasis was placed upon finding process conditions (e.g., a high CO overpressure) to prevent Pu volatilization from the kernel during high-temperature sintering and the early stages of TRISO coating.

Ammonia-based precipitation processes developed more recently, referred to as either "internal gelation" or "external gelation", have rendered the traditional sol-gel processes obsolete. Both internal and external gelation processes use a concentrated heavy-metal nitrate feed that is "acid-deficient", and therefore require a concentration/denitration solution preparation step. These new gel-precipitation methods are simpler, more reliable, and more suitable for scale-up and adaptation to radiochemical processing than the traditional sol-gel techniques.

Both internal and external gelation processes have undergone extensive development for the production of microspheres containing UO_2, UCO, ThO_2, and mixed systems of U-Th, U-Pu, and Th-Pu as well as pure carbides and nitrides. Only limited work has been done on pure Pu or Pu-rich systems. The limited internal gelation trials that have been conducted on feeds containing only plutonium revealed no fundamental barriers, but flowsheet optimization will be needed to avoid conditions that lead to cracking of the microspheres (Ref. 3). No significant complications are expected from inclusion of 5% Np in this process.

An external gelation/precipitation process has been used to manufacture both UCO and UO_2 kernels at GA in the US, at Nuclear Fuel Industries in Japan, and at HOBEG Company in Germany. The feasibility of producing 350 μm and 500 μm diameter UCO by this method has been demonstrated, although the quality requirements were not completely demonstrated for UCO material.

HEU UCO kernels of 195 μm diameter were fabricated in laboratory scale equipment by an internal gelation/precipitation process under the NP-MHTGR program. That material showed more uniform structure and density than similar kernels from external gelation, and on that basis the internal gelation process was selected as the reference process.

1.3 Data Needed

The existing internal gelation/precipitation process must be optimized, scaled up and qualified to assure that it is capable of manufacturing $PuO_{1.7}/NpO_x$ kernels which comply with the BD-MHR fuel product specifications. Data are needed for the completion of process specifications, the design of process equipment, the demonstration of product quality when fabricated in full size equipment, and for the generation of cost information for estimation of fuel manufacturing costs in a later fuel fabrication facility. The processes, equipment and procedures developed for DF kernel fabrication must account for the considerable radiological hazards posed by the Pu/Np feedstock. All process equipment shall be designed to be critically safe by geometry.

An important part of the data needed is the correlation between the key product attributes of density, kernel size, sphericity, stoichiometry, and levels of chemical impurities and the key process variables of feed chemical compositions, batch sizes, unit temperatures, unit pressures, reaction times, and flow rates. The range of process variables for which sensitivities are needed will be defined in a process development plan. Product quality must be established within a 95% confidence level by a statistical process control method.

Special attention must be given to the preparation of an acid-deficient feed solution because of the tendency of plutonium to hydrolyze and polymerize at low-acid conditions. Special laboratory methods (film evaporation at low temperature and pressure to avoid internal reflux) have been devised to prevent

Pu-polymerization during the concentration/denitration step. These methods do not appear suitable for scale-up, as they require evaporation of the metal nitrate solution to a solid to drive off most of the free acid. New methods which are more suitable for plant-scale operation must be developed. The design of the sintering apparatus will also require special attention because of the critical importance of the kernel stoichiometry. Basic refinements in process steps, such as microwave-induced gelation (Ref. 4) and continuous aging and washing (Ref. 5), shall be pursued for the purpose of improving process simplicity, reproducibility, and reliability. These improvements are a critical factor in establishing a workable radiochemical process for glovebox or hot-cell operations.

The manufacturing cost basis must be established for kernels to support program projections on fuel cycle costs. The basis for cost estimates will be data from operational campaigns in full scale equipment. Operational test data needed for cost estimates are batch sizes, times, yields of acceptable product and quantities of fuel material recovered from scrap for further processing. Facility data needed are the size of staff to perform all functions, the quantities of input materials and the capital costs of all equipment units.

1.4 Data Parameters/Service Conditions

The $PuO_{1.7}/NpO_x$ kernel production process must demonstrate the manufacture of kernels with the following product attributes and throughput goals:

Parameter	Value
Nominal diameter	[300] μm[1]
Nominal density	>10 Mg/m^3
Nominal composition	95% $PuO_{1.7}$/5% NpO_x
Oxygen-to-metal ratio	<1.7
Enrichment of Pu-239	[TBD] %[2]
Unit throughput	≥ [TBD] kg HM/day-module
Yield	≥[90]%

More detailed process conditions shall be developed to assure meeting the product requirements as defined in applicable product specifications (e.g., Ref. 6). Sensitivity tests shall be conducted to determine the effect of key process parameters on the product attributes. These tests shall include varying key process parameters over a range of conditions necessary to establish operating envelopes that support producing material that will meet the fuel specification.

Chemical impurity limits shall be specified for all feed materials and process chemicals introduced into the process. The introduction of chemical impurities shall be limited in all drying, heat-treatment and transfer unit operations so that the chemical impurities specified in the fuel product specifications will be met.

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[1] Values in [square brackets] are tentative and subject to change as the program requirements are refined.
[2] TBD = To be determined.

2. DESIGNER'S ALTERNATIVES

Alternatives to the proposed approach are as follows:

2.1 Use the external gelation process used previously in the US and Germany (by HOBEG) for U and Th kernels.

2.2 Use the powder agglomeration process for PuO_x/C kernel fabrication process used by Belgo Nucleaire for the production of Pu kernels irradiated in Dragon, or use a process that produces an oxycarbide kernel similar to that used for making the Dragon kernels, but based on an internal or external gelation method.

2.3 Use a sintering operation based on CO rather than H_2 to achieve substoichiometric plutonia, $PuO_{1.7}/NpO_x$.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected approach is to specify, construct and demonstrate the process flow sequence and the equipment units to fabricate $PuO_{1.7}/NpO_x$ kernels using the internal gelation process developed at ORNL.

Alternative 2.1 was not selected because previous test fuel from the external gelation process showed excessive variability of kernel density and stoichiometry and the presence of internal voids when used to produced HEU UCO. There is no experience base for using external gelation to produce pure plutonia, and the kinetics of plutonium precipitation are slower than uranium and thus may present a problem for the mass-transfer controlled external gelation process.

Alternatives 2.2 and 2.3 were not selected because the experience base for production of substoichiometric oxides by hydrogen firing of plutonium is greater than that for making oxycarbides, and H_2-based reduction has a greater experience base than CO-based reduction.

4. SCHEDULE REQUIREMENTS

Equipment meeting the requirements on confidence level for process control shall be in operation [6] months prior to the start of qualification irradiation testing.

5. PRIORITY

Urgency: TBD
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

The fallback position would be to use the external gelation process as partially developed previously and to pursue CO-based reduction of plutonia if development work indicates it is necessary as an improvement over the hydrogen firing baseline.

7. REFERENCES

1. Sanders, C.F., and J.D. Sease, “Fabrication and Characterization of Plutonium Test Element FTE-13: An HTGR Test Element Containing PuO_{2-x}, $Th_{0.75}Pu_{0.25}O_{2-x}$ and ThO_2,” ORNL-TM-4207, ORNL, August 1973.

2. “Development of Plutonium Fuelled Coated Particles for Irradiation in the Dragon Reactor Experiment,” Dragon Project Report DP-543, Belgo Nucleaire, May 1968.

3. Forthmanm, R. and Blass, G., “Fabrication of Uranium-Plutonium Oxide Microspheres by the Hydrolysis Process”, J. Nucl. Mat. **64** (1977), 275-280.

4. Yamagishi, S., “A new internal gelation process for fuel microsphere preparation without cooling initial solutions”, J. Nucl. Mat. **254** (1): 14-21 March 1998.

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6. "Fuel Product Specification for GT-MHR," DOE-GT-MHR 100209, Rev. 0, General Atomics, May 1994.

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

"AAA" Fuel Program
TRANSMUTATION FUEL KERNEL PROCESS DEVELOPMENT
DDN RTW.01.02

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

It is anticipated that the transmutation fuel kernels (Pu/Np/Am/Cm) will be manufactured by loading the heavy-metals on to an ion-exchange resin followed by high-temperature sintering to produce a kernel stoichiometry which is thermally stable during TRISO coating and under irradiation. Gelation/precipitation processes may also be considered as alternatives. Candidate processes, which have been developed on a laboratory scale, must be optimized, scaled up and demonstrated in full size equipment units to be certain that scaling effects are properly treated. The process details, including yields and costs, must be quantified for later transition to a commercial status. The scope of this DDN is to include the operations necessary for transformation of a separations plant product stream into a sintered microsphere suitable for TRISO coating. Solution preparation steps, such as concentration, denitration, and neutralization adjustment, are part of this process.

1.1 Summary of Functions/Assumptions

"Retain Radionuclides in Fuel Particles," Assumptions: Processes are available for manufacturing oxide-based fuel kernels for high integrity coated fuel particles.

1.2 Current Data Base Summary

A weak-acid resin (WAR) process was developed by GA and ORNL for the production of UC_xO_y and UC_2 kernels (Ref. 1). Uranyl ions (UO_2^{++}) were loaded on to a weak-acid resin in its ammonium form and subsequently dried, calcined, and converted to the desired stoichiometry. The resulting product was a porous ~300-μm kernel which proved to have a fission gas release rate several times higher than fully dense kernels. Because of this high gas release and heavy-metal loading limitations, development of WAR kernels was terminated in 1977.

A strong-acid resin process was developed at ORNL for making Am-Cm oxide kernels for use in transmutation targets for heavy element production in an aluminum based CERMET (Ref. 2). This process has been in operation for over thirty years at a lab-scale, and has been proven suitable for remote operations. Modifications of this process to make larger particles of a different chemistry would be necessary to support the fabrication of TF.

A traditional sol-gel process (developed at ORNL) was used to manufacture the PuO_{2-x} and $(Th,Pu)O_{2-x}$ kernels that were irradiated in Peach Bottom fuel test element FTE-13 (Ref. 3). The substoichiometric oxygen-to-plutonium ratios were obtained by heat treating in pure hydrogen at 1750 ^{o}C.

Belgo Nucleaire manufactured 500-μm PuO_x kernels which were diluted with carbon (C/PuO_2 = ~20) and subsequently TRISO coated for irradiation in the Dragon HTR and Studsvik (e.g., Ref. 4). The kernels were prepared by both sol gel and powder agglomeration processes. Considerable emphasis was placed upon finding process conditions (e.g., a high CO overpressure) to prevent Pu volatilization from the kernel during high-temperature sintering and the early stages of TRISO coating.

Ammonia-based precipitation processes developed more recently, referred to as either "internal gelation" or "external gelation", have rendered the traditional sol-gel processes obsolete. Both internal and external gelation processes use a concentrated heavy-metal nitrate feed that is "acid-deficient," and therefore require a concentration/denitration solution preparation step. These new gel-precipitation methods are simpler, more reliable, and more suitable for scale-up and adaptation to radiochemical processing than the traditional sol-gel techniques. Both internal and external gelation processes have seen extensive development for the production of microspheres containing, UO_2, UCO, ThO_2, and mixed systems U-Th, U-Pu, Th-Pu, as well as pure carbides and nitrides.

An external gelation/precipitation process has been used to manufacture both UCO and UO_2 kernels at GA in the US, at Nuclear Fuel Industries in Japan, and at HOBEG Company in Germany. The feasibility of producing 350 μm and 500 μm diameter UCO by this method has been demonstrated, although the quality requirements were not completely demonstrated for UCO material.

1.3 Data Needed

A process must be developed, qualified, and scaled up to assure that it is capable of manufacturing TF kernels which comply with the fuel product specifications. It is anticipated that the TF kernels (Pu/Np/Am/Cm) will be manufactured by loading the heavy-metals on to an ion-exchange resin followed by high-temperature sintering to produce a kernel stoichiometry which is thermally stable during TRISO coating and under irradiation. A particular concern is the potential volatilization and loss of americium from the kernel during high-temperature sintering and subsequent TRISO coating. Because of this concern, consideration of inorganic ion-exchangers/sorbents will be included in addition to the established organic ion-exchange resins. Gelation/precipitation processes may also be considered as alternatives.

Data are needed for the completion of process specifications, the design of process equipment, the demonstration of product quality when fabricated in full size equipment, and cost information for estimation of fuel manufacturing costs in a later fuel fabrication facility. The processes, equipment and procedures developed for DF kernel fabrication must account for the considerable radiological hazard and radiolysis potential of the Pu/Np/Am/Cm feedstock. All process equipment shall be designed to be critically safe by geometry.

An important part of the data needed is the correlation between the key product attributes of density, kernel size, sphericity, stoichiometry, and levels of chemical impurities to the key process variables of feed chemical compositions, batch sizes, unit temperatures, unit pressures, reaction times, and flow rates. The range of process variables for which sensitivities are needed will be defined in a process development plan. Product quality must be established within a 95% confidence level by a statistical process control method.

Various strategies are available for distribution of the actinides in the kernel, and these will need to be evaluated on the basis of their thermochemical viability (e.g., oxygen management, Am volatility) and fabrication practicality. For example, homogeneous kernels containing a uniform mix of the actinides can be made most easily by gel-precipitation methods. Homogenous kernels from a resin-loading process are readily obtained only from chemically similar groups of elements. The Np/Pu fraction and the Am/Cm fraction are likely to be received as separate streams from the separations plant, and will naturally load as a group (Np/Pu together and Am/Cm together) on the ion-exchanger. A homogeneous loading of these two disparate groups on ion-exchangers is atypical. If a homogeneous kernel is required, then methods to obtain a homogenous mix of Np-Pu-Am-Cm by resin-loading will need to be devised and evaluated for its suitability to remote operations.

1.4 Data Parameters/Service Conditions

The TF kernel production process must demonstrate the manufacture of kernels with the following product attributes and throughput goals:

Parameter	Value
Nominal Composition (atom %)	
Pu	65
Np	5
Am	25
Cm	5
Nominal diameter	[200] µm
Nominal density	>10 Mg/m^3
Oxygen-to-metal ratio	< [TBD]
Enrichment of Pu-239	[TBD] %
Unit throughput	≥ [TBD] kg HM/day-module
Yield	≥[90]%

More detailed process conditions shall be developed to assure meeting the product requirements as defined in applicable product specifications (e.g., Ref. 5). Sensitivity tests shall be conducted to determine the effect of key process parameters on the product attributes. These tests shall include varying the key process parameters over the range of conditions necessary to establish operating envelopes that support producing material that will meet the fuel specification.

Chemical impurity limits shall be specified for all feed materials and process chemicals introduced into the process. The introduction of chemical impurities shall be limited in all drying, heat-treatment and transfer unit operations so that the chemical impurities specified in the fuel product specifications will be met.

2. DESIGNER'S ALTERNATIVES

Alternatives to the proposed approach are as follows:

2.1 Use a gel-precipitation process, either internal or external gelation.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected approach is to specify, construct and demonstrate the process flow sequence and the equipment units needed to fabricate TF kernels. It is anticipated that the TF kernels will be manufactured by loading the heavy-metals on to an ion-exchange resin followed by high-temperature sintering to produce a kernel stoichiometry which is thermally stable during TRISO coating and under irradiation. The simplicity and proven applicability of this process to hot-cell operations is a primary factor in its selection.

Alternative 2.1 was not selected because of the greater complexity of this process for adaptation to hot-cell operations.

4. SCHEDULE REQUIREMENTS

Equipment meeting the requirements on confidence level for process control shall be in operation three months prior to the start of qualification irradiation testing.

5. PRIORITY

Urgency: TBD
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

The fallback position would be to investigate the production of TF kernels by gel-precipitation processes a or powder agglomeration process.

The ultimate fallback position would be to eliminate the production and irradiation of transmutation fuel from the proposed fuel cycle and to send to the high-burnup, spent driver fuel directly to a geological repository. The programmatic consequences are unknown but are anticipated to be significant.

7. REFERENCES

1. Dobson, J.C., and L.C. Brown, "Fissile Fuel Kernel Production Technology," GA Document 904401, Rev. A, General Atomic, August, 1980.

2. F. R. Chattin et al., "Preparation of Curium-Americium Oxide Microspheres by Resin-Bead Loading," in ACS Symposium Series, No. 161: Transplutonium Elements-Production and Recovery, J. D. Navratil and W. W. Schulz, eds., American Chemical Society, pp. 173-185 (1981).

3. Sanders, C.F., and J.D. Sease, “Fabrication and Characterization of Plutonium Test Element FTE-13: An HTGR Test Element Containing PuO_{2-x}, $Th_{0.75}Pu_{0.25}O_{2-x}$ and ThO_2,” ORNL-TM-4207, August 1973.

4. “Development of Plutonium Fuelled Coated Particles for Irradiation in the Dragon Reactor Experiment,” Dragon Project Report DP-543, Belgo Nucleaire, May 1968.

5. "Fuel Product Specification for GT-MHR," document No. DOE-GT-MHR 100209, Rev. 0, May 1994.

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

“AAA” Fuel Program
DF PARTICLE COATING PROCESS DEVELOPMENT
DDN RTW.01.03

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The fuel quality and performance goals for the DB-MHR Driver Fuel require that the as-manufactured, heavy-metal contamination and coated particle defects levels and the in-reactor failure fractions be reduced relative to demonstrated US experience for TRISO-coated fuels. Coating process improvements are essential for producing reactor fuel having the as-manufactured coating attributes mandated by the DB-MHR fuel product specification and performance requirements.

1.1 Summary of Function/Assumptions

"Retain Radionuclides in Fuel Particles with TRISO Coatings," Assumption: Processes are available for manufacturing coated fuel particles that maintain high integrity under irradiation and accident conditions.

1.2 Current Data Base Summary

TRISO coatings have been applied to various fuel kernels, including PuO_{2-x}, $(Th,Pu)O_{2-x}$, PuO_x/C, and NpO kernels, in the US, Europe, Japan and China.

The basic principles for applying TRISO coatings on dense microspheres by chemical vapor deposition are well established after nearly four decades of international development. The exhaustive study by Voice in the 1960s of the relationships between coating process parameters and the attendant physical properties of the SiC coating remains a seminal work (Ref. 1). More recent TRISO coating process experience in Germany and the United Kingdom has also been described (Refs 2 and 3, respectively). Nevertheless, in certain regards, the production of high-quality coatings, especially PyC coatings, is as much art as science; for example, the properties of the coatings can be affected by the mechanical design of the coater.

The PuO_{2-x} and $(Th,Pu)O_{2-x}$ kernels irradiated in Peach Bottom fuel test element FTE-13 were TRISO-coated in a 1□-in.-diameter, single-inlet coater at ORNL (Ref. 4). The pyrocarbon coatings were conventional coatings: the buffer layer was derived from acetylene, and the low-temperature isotropic (LTI) IPyC and OPyC coatings were derived from propylene. The SiC coatings were deposited at 1500 – 1550 ^{o}C from methyltrichlorosilane with a hydrogen carrier gas.

The PuO_x/C kernels irradiated in the Dragon HTR and Studsvik were TRISO-coated in a small laboratory coater. The LTI carbon coatings were deposited from acetylene at 1500 °C, and the SiC coatings were deposited at 1450 – 1500 °C from methyltrichlorosilane with a hydrogen carrier gas (Ref. 5). Different coating conditions were investigated with the objective of minimizing potential plutonium volatilization from the kernel during coating.

General Atomics mass produced TRISO-coated HEU $(Th,U)C_2/ThC_2$ fuel (33,000 kg HM) by for the Fort St. Vrain HTGR (Ref. 6), but the as-manufactured HM contamination and coating defect fractions were at least an order-of-magnitude higher than that required for the DB-MHR reactor. The German company HOBEG mass produced TRISO-coated LEU UO_2 reload fuel for the AVR which met the quality requirements tentatively adopted for the DB-MHR reactor.

1.3 Data Needed

Product specifications will be written to define the requisite attributes of TRISO fuel capable of meeting the DB-MHR performance requirements. The TRISO coating process must be scaled up and qualified to assure that it is capable of manufacturing DF particles which comply with the fuel product specifications (Ref. 7). Data are needed for the completion of process specifications, the design of process equipment, the demonstration of product quality when fabricated in full size units, and cost information for estimation of fuel manufacturing costs in a later fuel fabrication facility. The processes, equipment and procedures developed for TRISO coating DF kernels must account for the considerable radiological hazards posed by the Pu/Np kernels. All process equipment shall be designed to be critically safe by geometry.

Test data from quality control measurements of coated particle batches are needed to qualify the product for obtained with known process control conditions. The process equipment and specifications must be developed to such an extent that the sensitivities of the product attributes to the key process variables are accounted for and acceptable boundaries defined quantitatively.

The key process parameters to be controlled are the coater configuration, batch size, coating temperature, inlet gas composition, gas mass flow rate, and coating time for each coating layer. The product attributes are summarized in Section 1.4. The range of process parameter variances for which sensitivities are needed will be defined in a process development plan. The effects of chemical impurities needs to be quantified for all feed materials and process chemicals introduced into the process. The introduction of chemical impurities shall be limited in all coating and transfer unit operations so that the chemical impurity limits defined in the fuel product specifications will be met. Data are also needed to determine if the TRISO coating layers must be deposited in a continuous series of operations, without the cool down and unloading between layers that has been the previous practice in the U.S.

1.4 Data Parameters/Service Conditions

The coating process, equipment, and procedures must produce TRISO coated particles which meet the acceptance criteria for properties and dimensions as defined in the fuel product specification for the DB-MHR DF particles. Conditions for coater unit operations will be elaborated in fuel process specifications.

The coating process must be capable of fabricating fuel meeting the following product attributes:

Design Parameter	Value	
Kernel material	95% $PuO_{1.7}$/5% NpO	
Enrichment, % Pu-239:	[TBD]	
Kernel		
Diameter, µm:	[300 ± 10]	
Density, kg/m^3	>10	
Buffer		
Thickness, µm:	[150 ± 10]	
Density, kg/m^3	1	
Inner PyC		
Thickness, µm:	35 ± 5	
Density, kg/m^3	1.85	
SiC		
Thickness, µm:	35 ± 5	
Density, kg/m^3	3.2	
Outer PyC		
Thickness, µm	40 ± 10	
Density, kg/m^3	1.85	
Coating Defects	≥50% Confidence	≥95% Confidence
Missing buffer	[$\leq 1.0 \times 10^{-5}$]	[$\leq 2.0 \times 10^{-5}$]
SiC defects	[$\leq 5.0 \times 10^{-5}$]	[$\leq 1.0 \times 10^{-4}$]
Batch Size, kg/charge	≥ [TBD]	
Yield	≥[90]%	

More detailed process conditions shall be developed to assure meeting the product requirements as defined in applicable specifications. Sensitivity tests shall be conducted to determine the effect of key process parameters on the product attributes. These tests shall include varying the key process parameters over the range of conditions necessary to establish operating envelopes that support producing material that will meet the fuel specification.

2. DESIGNER'S ALTERNATIVES

Alternatives to the proposed approach are as follows:

2.1. Scale down the standard German coater to a critically safe geometry, and use German process conditions.

2.2. Use Fort St. Vrain coating process technology and operating parameters.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected design approach is to take the basic German coater design and process conditions, which were developed and qualified for uranium and thorium fuels, and optimize them for coating [300] μm Pu/Np kernels. Changes will be made in the coater design and operating procedures to facilitate remote operations and maintenance as required for the processing of Pu fuels.

Alternative 2.1 was not chosen because the standard German coater design is not conducive to remote operations and maintenance. It is anticipated that the German process conditions will prove satisfactory for coating Pu/Np kernels.

Alternative 2.2 was not chosen because the contamination levels and the SiC defect fractions expected to be produced with the FSV coater technology would be greater than allowed by the fuel product specifications.

4. SCHEDULE REQUIREMENTS

Coated DF particles shall be required for screening or developmental capsule irradiation tests by [TBD]. Coating process conditions and process specifications shall be established documented and approved by the ORNL program manager [three] months prior to the start of fuel manufacturing for qualification irradiation tests.

5. PRIORITY

Urgency: TBD
Cost benefit: H
Uncertainty in existing data: M
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

The fallback position would be to scale down German coater (Alternative 2.1) and develop alternative O&M procedures that could be remotized.

7. REFERENCES

1. Voice, E.H., and D.N. Lamb, "The Deposition and Structure of Pyrolytic Silicon Carbide," Dragon Project Report DP-667, O.E.C.D High Temperature Reactor Project, 1969.

2. Huschka, H., and P. Vygen, "Coated Fuel Particles: Requirements and Status of Fabrication Technology", Nuclear Technology, Vol. 35, No. 2, September 1977.

3. Lefevre, R.L.R., and M.S.T. Price, "Coated Nuclear Fuel Particles: The Coating Process and Its Model", Nuclear Technology, Vol. 35, No. 2, September 1977.

4. Sanders, C.F., and J.D. Sease, "Fabrication and Characterization of Plutonium Test Element FTE-13: An HTGR Test Element Containing PuO_{2-x}, $Th_{0.75}Pu_{0.25}O_{2-x}$ and ThO_2," ORNL-TM-4207, August 1973.

5. Horsley, G.W., et al., "The Manufacture of Plutonium-Fuelled Fission-Product-Retaining Coated Particles for Irradiation in the Dragon Reactor Experiment," Dragon Project Report DP-525.

6. "MHTGR Fuel Process and Quality Control Description," document No. DOE HTGR-90257, September 1991.

7. "Fuel Product Specification For GT-MHR," document No. DOE-GT-MHR-100209, Rev. 0, May 1994.

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

“AAA” Fuel Program
TF PARTICLE COATING PROCESS DEVELOPMENT
DDN RTW.01.04

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The fuel quality and performance goals for the DB-MHR Transmutation Fuel require that the as-manufactured, heavy-metal contamination and coated particle defects levels and the in-reactor failure fractions be reduced relative to demonstrated US experience for TRISO-coated fuels. Coating process improvements are essential for producing reactor fuel having the as-manufactured coating attributes mandated by the fuel product specification and DB-MHR performance requirements.

1.1 Summary of Function/Assumptions

"Retain Radionuclides in Fuel Particles with TRISO Coatings," Assumption: Processes are available for manufacturing coated fuel particles that maintain high integrity under irradiation and accident conditions.

1.2 Current Data Base Summary

TRISO coatings have been applied to various fuel kernels, including PuO_{2-x}, $(Th,Pu)O_{2-x}$, PuO_x/C, and NpO kernels, in the US, Europe, Japan and China.

The basic principles for applying TRISO coatings on dense microspheres by chemical vapor deposition are well established after nearly four decades of international development. The exhaustive study by Voice in the 1960s of the relationships between coating process parameters and the attendant physical properties of the SiC coating remains a seminal work (Ref. 1). More recent TRISO coating process experience in Germany and the United Kingdom has also been described (Refs 2 and 3, respectively). Nevertheless, in certain regards, the production of high-quality coatings, especially PyC coatings, is as much art as science; for example, the properties of the coatings can be affected by the mechanical design of the coater.

The PuO_{2-x} and $(Th,Pu)O_{2-x}$ kernels irradiated in Peach Bottom fuel test element FTE-13 were TRISO-coated in a 1▯-in.-diameter, single-inlet coater at ORNL (Ref. 4). The pyrocarbon coatings were conventional coatings: the buffer layer was derived from acetylene, and the low-temperature isotropic (LTI) IPyC and OPyC coatings were derived from propylene. The SiC coatings were deposited at 1500 – 1550 °C from methyltrichlorosilane with a hydrogen carrier gas.

The PuO_x/C kernels irradiated in the Dragon HTR and Studsvik were TRISO-coated in a small laboratory coater. The LTI pyrocarbon coatings were deposited from acetylene at 1500 ^{o}C, and the SiC coatings were deposited at 1450 – 1500 ^{o}C from methyltrichlorosilane with a hydrogen carrier gas (Ref. 5). Different coating conditions were investigated with the objective of minimizing potential plutonium volatilization from the kernel during coating.

Conventional TRISO coatings have been successfully applied to small quantities of ^{237}NpO kernels in a laboratory-scale coater at General Atomics. TRISO coatings have (apparently) not been applied to kernels with a high americium content.

General Atomics mass produced TRISO-coated HEU $(Th,U)C_2/ThC_2$ fuel (33,000 kg HM) by for the Fort St. Vrain HTGR (Ref. 6), but the as-manufactured HM contamination and coating defect fractions were at least an order-of-magnitude that that required for the DB-MHR reactor. The German company HOBEG mass produced TRISO-coated LEU UO_2 reload fuel for the AVR which met the quality requirements tentatively adopted for the DB-MHR reactor.

1.3 Data Needed

Product specifications will be written to define the requisite attributes of TRISO fuel capable of meeting the DB-MHR performance requirements. The TRISO coating process must be scaled up and qualified to assure that it is capable of manufacturing TF particles which comply with the fuel product specifications (Ref. 7). A particular concern is the potential volatilization and loss of americium from the kernel during the early stages of coating. Data are needed for the definition of process specifications, the design of process equipment, the demonstration of product quality when fabricated in full size units, and cost information for estimation of fuel manufacturing costs in a later fuel fabrication facility. The processes, equipment and procedures developed for TRISO coating TF kernels must account for the considerable radiological hazards posed by the Pu/Np/Am/Cm kernels. All process equipment shall be designed to be critically safe by geometry.

Test data from quality control measurements of coated particle batches are needed to qualify the product obtained with known process conditions. The process equipment and specifications must be developed to such an extent that the sensitivities of the product attributes to the key process variables are determined and acceptable envelopes are defined quantitatively.

The key process parameters to be controlled are: coater configuration, batch size, coating temperature, inlet gas composition, mass flow rate, and coating time for each coating layer. The product attributes are summarized in Section 1.4. The range of process parameter variances for which sensitivities are needed will be defined in a process development plan. The effects of chemical impurities need to be quantified for all feed materials and process chemicals introduced into the process. The introduction of chemical impurities shall be limited in all coating and transfer unit operations so that the chemical impurity limits specified in the fuel product specifications will be met. Data are also needed to determine if the TRISO coating layers must be deposited in a continuous series of operations, without the cool down and unloading between layers that have been the previous practice in the U.S.

1.4 Data Parameters/Service Conditions

The coating process, equipment, and procedures must produce TRISO coated particles which meet the acceptance criteria for properties and dimensions as defined in the fuel product specification for the DB-MHR TF particles. Conditions for coater unit operations will be elaborated in fuel process specifications.

The coating process must be capable of fabricating TF particles meeting the following product attributes:

Design Parameter	Value
Nominal Composition (atom %)	
Pu	65
Np	5
Am	25
Cm	5
Enrichment, % Pu-239:	[TBD]
Kernel	
Diameter, µm:	[200 ± 10]
Density, kg/m^3	>10
Buffer	
Thickness, µm:	>100
Density, kg/m^3	1
Inner PyC	
Thickness, µm:	35 ± 5
Density, kg/m^3	1.85
SiC	
Thickness, µm:	35 ± 5
Density, kg/m^3	3.2
Outer PyC	
Thickness, µm	40 ± 10
Density, kg/m^3	1.85

Coating Defects	≥50% Confidence	≥95% Confidence
Missing buffer	[$\leq 1.0 \times 10^{-5}$]	[$\leq 5.0 \times 10^{-5}$]
SiC defects	[$\leq 5.0 \times 10^{-5}$]	[$\leq 2.5 \times 10^{-4}$]
Batch Size, kg/charge	≥ [TBD]	
Yield	≥[90]%	

More detailed process conditions shall be developed to assure meeting the product requirements as defined in applicable product specifications and tests specifications. Sensitivity tests shall be conducted to determine the effect of key process parameters on the product attributes. These tests shall include varying the key process parameters over the range of conditions necessary to establish operating envelopes that support producing material that will meet the fuel specification.

2. DESIGNER'S ALTERNATIVES

Alternatives to the proposed approach are as follows:

2.1. Scale down the standard German coater to a critically safe geometry and use German process conditions.

2.2. Use Fort St. Vrain coating process technology and operating parameters.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected design approach is to take the basic German coater design, which was developed and qualified for uranium and thorium fuels, and optimize it for coating [200] µm Pu/Np/Am/Cm kernels. Changes will be made in the coater design and operating procedures to facilitate remote operations and maintenance as required for the processing of Pu/Np/Am/Cm fuels. German process conditions will be the point of departure, but changes in the process conditions and/or in the coating design may be necessary to minimize the potential volatilization and loss of americium from the kernel during the early stages of coating.

Alternative 2.1 was not chosen because the standard German coater design is not conducive to remote operations and maintenance. In addition, changes in the German process conditions may be necessary to minimize the potential volatilization and loss of americium from the kernel during coating.

Alternative 2.2 was not chosen because the contamination levels and the SiC defect fractions expected to be produced with the FSV coater technology would be greater than allowed by the fuel product specifications.

4. SCHEDULE REQUIREMENTS

Coated TF particles shall be required for screening or developmental capsule irradiation tests by [TBD]. Coating process conditions and process specifications shall be documented and approved by the ORNL program manager [three] months prior to the start of fuel manufacturing for qualification irradiation tests.

5. PRIORITY

Urgency: TBD
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

The fallback position would be to scale down the German coater (Alternative 2.1) and develop alternative O&M procedures that could be remotized. The question of whether or not the use of German process conditions would lead unacceptable Am losses from the kernel can not be answered until some scoping coating runs have been made with prototypical kernels. Alternatively, it may be that if a low-temperature seal coat (PyC or ZrC) is first applied to the TF kernel, the German process conditions can be used for the rest of the coating system.

7. REFERENCES

1. Voice, E.H., and D.N. Lamb, “The Deposition and Structure of Pyrolytic Silicon Carbide,” Dragon Project Report DP-667, O.E.C.D High Temperature Reactor Project, 1969.

2. Huschka, H., and P. Vygen, "Coated Fuel Particles: Requirements and Status of Fabrication Technology", Nuclear Technology, Vol. 35, No. 2, September 1977.

3. Lefevre, R.L.R., and M.S.T. Price, "Coated Nuclear Fuel Particles: The Coating Process and Its Model", Nuclear Technology, Vol. 35, No. 2, September 1977.

4. Sanders, C.F., and J.D. Sease, “Fabrication and Characterization of Plutonium Test Element FTE-13: An HTGR Test Element Containing PuO_{2-x}, $Th_{0.75}Pu_{0.25}O_{2-x}$ and ThO_2,” ORNL-TM-4207, August 1973.

5. Horsley, G.W., et al., “The Manufacture of Plutonium-Fuelled Fission-Product-Retaining Coated Particles for Irradiation in the Dragon Reactor Experiment,” Dragon Project Report DP-525.

6. "MHTGR Fuel Process and Quality Control Description," document No. DOE HTGR-90257, September 1991.

7. "Fuel Product Specification For GT-MHR," document No. DOE-GT-MHR-100209, Rev. 0, May 1994.

Originator	Date
Engineering Manager	Date
Project Manager	Date

DATE: 09/27/02

"AAA" Fuel Program
DF COMPACT FABRICATION PROCESS
DDN RTW.01.05

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The fuel compact provides a bonding medium for confining the coated fuel particles within a defined body during assembly and reactor operation. Process development is required to establish the process conditions for fabricating DF compacts which meet all product specifications, including low defects in fuel elements.

1.1 Summary of Functions/Assumptions

"Control Transport of Radionuclides from Core," Assumption: Processes are available for manufacturing high quality fuel compacts for inclusion in prismatic fuel elements.

1.2 Current Data Base Summary

Extensive experience in the production of cylindrical fuel compacts has been gained at GA during production of Fort St. Vrain (FSV) fuel and through later process development for the MHTGR (Ref. 1). Similar experience has been demonstrated for spherical fuel elements with overcoated fuel particles for the German pebble bed reactor (Ref. 2). Annular fuel compacts using a similar overcoating process has been developed in Japan (Ref. 3).

Two different processes were used to fabricate the fuel compacts containing the TRISO-coated PuO_{2-x} and $(Th,Pu)O_{2-x}$ particles irradiated in Peach Bottom fuel test element FTE-13 (Ref. 4). Most of the fuel compacts were made with a slug-injection process, but those compacts containing the smallest diameter particles were fabricated by the Admix process. All of the compacts were fabricated with the FSV petroleum-pitch matrix, and all were heat treated in argon at 1800 ^{o}C for 30 min. The TRISO-coated PuO_x/C particles irradiated in the Dragon HTR and Studsvik were irradiated as loose particle beds in graphite holders.

The product specifications for FSV fuel were much less restrictive than the provisional requirements for the DB-MHR core. While much of the prior experience is of value to the program, the process must be improved to reduce the level of heavy-metal contamination and defective particles in DF compacts by more than an order of magnitude compared to the levels demonstrated during FSV mass production.

A compact process improvement program was conducted at General Atomics in 1995-1996 (Ref. 5). High-quality, German LEU UO_2 particles were used in the study to eliminate the particles as a potential source of as-manufactured heavy-metal contamination and coating defects in the finished compacts. Improvements were made in the control of the matrix-injection step which significantly reduced the mechanical stresses in the coated particles during the formation of green compacts. More importantly, these process studies determined that a major source of heavy-metal contamination and SiC coating defects were chemical impurities, especially transition metals, in the petroleum-pitch matrix and in the heat treatment furnace, principally iron, which attack the SiC coating during the high-temperature heat treatment step. Process changes were made to reduce the impurity levels by additional HCl leaching of the feedstocks, and the furnace internals were thoroughly cleaned. As a result of these improvements, fuel compacts (referred to as "MHR-1 compacts") were fabricated that met commercial GT-MHR product quality specifications with large margins.

1.3 Data Needed

Compacting processes based upon both resin matrix and petroleum pitch matrix shall be evaluated, and a reference process for fabricating DF compacts shall be chosen with the other process serving as backup. The selected compacting process must be scaled up and qualified to assure that it is capable of mass producing compacts which comply with the DF product specification (e.g., Ref. 6).

Data are needed for the completion of process specifications, the design of process equipment, the demonstration of product quality when fabricated in full size equipment units, and cost information for estimation of fuel manufacturing costs in a later fuel fabrication facility. The processes, equipment and procedures developed for DF compact fabrication must account for the considerable radiological hazards posed by the Pu/Np particles and for the heat generation from Pu decay. All process equipment shall be designed to be critically safe by geometry

A process flow sheet, compact equipment specifications and demonstrated equipment performance must be completed for all steps in the fuel compact process. The process parameters and the specifications must be developed to such an extent that the sensitivities of the compact attributes to variances of the key process parameters are known. Operational test data are needed on batch sizes, compacting times, yield of acceptable product and quantities of fuel material recovered from scrap for further processing. Documented product quality data are required, with the quality established within a 95% confidence level by statistical process control methods in qualification campaigns.

The key process parameters to be controlled are the batch sizes, matrix injection temperature and pressure, heat treatment temperature, levels of impurities in the feedstocks and process equipment, and times for each process step. The product attributes are described in Section 1.4.

1.4 Data Parameters/Service Conditions

Conditions for fuel compact fabrication will be defined in fuel process specifications. The compacting process must be capable of fabricating fuel meeting the following product attributes, as specified in the fuel product specification:

Design Parameter	Value	
Matrix composition	Phenolic resin Petroleum pitch	
Diameter, mm	12.45	
Length, mm	49.3	
Particle Packing fraction	≤[TBD]	
Defect Fractions in Compacts	≥50% Confidence	≥95% Confidence
HM Contamination	[≤1.0 x 10^{-5}]	[≤2.0 x 10^{-5}]
SiC defects	[≤5.0 x 10^{-5}]	[≤1.0 x 10^{-4}]
HM Contamination + SiC Defects	[≤6.0 x 10^{-5}]	[≤1.2 x 10^{-4}]

More detailed product requirements and process conditions shall be defined in applicable product specifications. Sensitivity tests shall be conducted to determine the effect of key process parameters on the product attributes. These tests shall include varying the key process parameters over the range of conditions necessary to establish operating envelopes that support producing material that will meet the fuel specification.

2. DESIGNER'S ALTERNATIVES

Alternatives to the proposed approach are as follows:

2.1. Utilize a variant of the German or Japanese concept with mechanical matrix overcoating of particles prior to compacting and a resin based matrix.

2.2. Utilize FSV process (petroleum pitch matrix), and accept fuel compacts with higher levels of damaged coated fuel particles. With this alternative, some modifications would be likely required to the reactor design with a need to accept higher levels of fission product release to the reactor primary coolant circuit.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected approach is to evaluate compacting processes based upon resin matrix with the petroleum pitch process as the backup process.

The alternative 2.1 approach was not selected because this concept has severe limitations on the volume available for coated fuel particles within the fuel compact (i.e., packing fraction), and this may unacceptably limit the power density and/or the fuel lifetime in the reactor core.

The alternative 2.2 approach was not selected because of the uncertainty in being able to meet the reactor radionuclide requirements without a pressurized secondary containment and active cooling systems, and the risk of delays at the time of construction of a fuel fabrication facility. In addition, the FSV process, especially the curing of green compacts in beds of Al_2O_3 powder, is ill suited for remote processing of coated Pu fuels.

4. SCHEDULE REQUIREMENTS

Fuel compacts shall be required for screening or developmental capsule irradiation tests by [TBD]. Fuel compact process conditions and process specifications shall be documented and approved by the ORNL program manager [three months] prior to the start of fuel manufacturing for qualification irradiation tests.

5. PRIORITY

Urgency: TBD
Cost Benefit: H
Uncertainty in existing data: M
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

The fallback position would be the alternative 2.2 approach (i.e., the FSV process). The consequences of nonexecution would be a risk of inadequate performance of qualification fuel, the potential for a requirement to add a pressurized secondary containment to the reactor plant, and potential schedule delays in demonstrating the fuel for a first reactor plant.

7. REFERENCES

1. "MHTGR Fuel Process and Quality Control Description," DOE-HTGR-90257, September 1991.

2. Nabielek, H., et al., "Development of Advanced HTR Fuel Elements," Nuclear Engineering & Design, 1990.

3. Saito, S., et al, "Safety Requirements and Research and Development on HTTR Fuel," Report IWG-GCR/25 International Atomic Energy Agency, 1991

4. Sanders, C.F., and J.D. Sease, “Fabrication and Characterization of Plutonium Test Element FTE-13: An HTGR Test Element Containing PuO_{2-x}, $Th_{0.75}Pu_{0.25}O_{2-x}$ and ThO_2,” ORNL-TM-4207, August 1973.

5. Goodin, D., “MHR-1 Compact Process Development Test Report,” GA Document 910906, Rev. 0, November 1996.

6. "Fuel Product Specification for GT-MHR," DOE-GT-MHR 100209, Rev. 0, May 1994.

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

"AAA" Fuel Program
TF COMPACT FABRICATION PROCESS
DDN RTW.01.06

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The fuel compact provides a bonding medium for confining the coated fuel particles within a defined body during assembly and reactor operation. The process development is required to establish the process conditions for fabricating TF compacts which meet all product specifications, including low defects in fuel assemblies.

1.1 Summary of Functions/Assumptions

"Control Transport of Radionuclides from Core," Assumption: Processes are available for manufacturing high quality fuel compacts for inclusion in prismatic fuel elements.

1.2 Current Data Base Summary

Extensive experience in the production of cylindrical fuel compacts has been gained at GA during production of Fort St. Vrain (FSV) fuel and through later process development for the MHTGR (Ref. 1). Similar experience has been demonstrated for spherical fuel elements with overcoated fuel particles for the German pebble bed reactor (Ref. 2). Annular fuel compacts using a similar overcoating process has been developed in Japan (Ref. 3).

Two different processes were used to fabricate the fuel compacts containing the TRISO-coated PuO_{2-x} and $(Th,Pu)O_{2-x}$ particles irradiated in Peach Bottom fuel test element FTE-13 (Ref. 4). Most of the fuel compacts were made with a slug-injection process, but those compacts containing the smallest diameter particles were fabricated by the Admix process. All of the compacts were fabricated with the FSV petroleum-pitch matrix, and all were heat treated in argon at 1800 °C for 30 min. The TRISO-coated PuO_x/C particles irradiated in the Dragon HTR and Studsvik were irradiated as loose particle beds in graphite holders.

The product specifications for FSV fuel were much less restrictive than the provisional requirements for the DB-MHR core. While much of the prior experience is of value to the program, the process must be improved to reduce the level of heavy-metal contamination and defective particles in TF compacts by more than an order of magnitude compared to the levels demonstrated during FSV mass production.

A compact process improvement program was conducted at General Atomics in 1995-1996 (Ref. 5). High-quality, German LEU UO_2 particles were used in the study to eliminate the particles as a potential source of as-manufactured heavy-metal contamination and coating defects in the finished compacts. Improvements were made in the control of the matrix-injection step which significantly reduced the mechanical stresses in the coated particles during the formation of green compacts. More importantly, these process studies determined that a major source of heavy-metal contamination and SiC coating defects were chemical impurities, especially transition metals, in the petroleum-pitch matrix and in the heat treatment furnace, principally iron, which attack the SiC coating during the high-temperature heat treatment step. Process changes were made to reduce the impurity levels by additional HCl leaching of the feedstocks, and the furnace internals were thoroughly cleaned. As a result of these improvements, fuel compacts (referred to as “MHR-1 compacts”) were fabricated that met commercial GT-MHR product quality specifications with large margins.

1.3 Data Needed

Compacting processes based upon both resin matrix and petroleum pitch matrix shall be evaluated, and a reference process for fabricating TF compacts shall be chosen with the other process serving as backup. The selected compacting process must be scaled up and qualified to assure that it is capable of mass producing compacts which comply with the TF product specification (e.g., Ref. 6).

Data are needed for the completion of process specifications, the design of process equipment, the demonstration of product quality when fabricated in full size equipment units, and cost information for estimation of fuel manufacturing costs in a later fuel fabrication facility. The processes, equipment and procedures developed for TF compact fabrication must account for the considerable radiological hazards posed by the Pu/Np/Am/Cm particles and for the heat generation from actinide decay. All process equipment shall be designed to be critically safe by geometry

A process flow sheet, compact equipment specifications and demonstrated equipment performance must be completed for all steps in the fuel compact process. The process parameters and the specifications must be developed to such an extent that the sensitivities of the compact attributes to variances of the key process parameters are known. Operational test data are needed on batch sizes, compacting times, yield of acceptable product and quantities of fuel material recovered from scrap for further processing. Documented product quality data are required, with the quality established within a 95% confidence level by statistical process control methods in qualification campaigns.

The key process parameters to be controlled are the batch sizes, matrix injection temperature and pressure, heat treatment temperature, levels of impurities in the feedstocks and process equipment, and times for each process step. The product attributes are described in Section 1.4.

1.4 Data Parameters/Service Conditions

Conditions for fuel compact fabrication will be defined in fuel process specifications. The compacting process must be capable of fabricating fuel meeting the following product attributes, as specified in the fuel product specification (e.g., Ref. 6):

Design Parameter		Value
Matrix composition		Phenolic resin binder Graphite shim particles Graphite powder filler
Diameter, mm		12.45
Length, mm		49.3
Particle Packing Fraction		≤[TBD]
Defect Fractions in Compacts	≥50% Confidence	≥95% Confidence
HM Contamination	[$\leq 1.0 \times 10^{-5}$]	[$\leq 5.0 \times 10^{-5}$]
SiC defects	[$\leq 5.0 \times 10^{-5}$]	[$\leq 2.5 \times 10^{-4}$]
HM Contamination + SiC Defects	[$\leq 6.0 \times 10^{-5}$]	[$\leq 3.0 \times 10^{-4}$]

More detailed product requirements and process conditions shall be defined in applicable product specifications. Sensitivity tests shall be conducted to determine the effect of key process parameters on the product attributes. These tests shall include varying the key process parameters over the range of conditions necessary to establish operating envelopes that support producing material that will meet the fuel specification.

2. DESIGNER'S ALTERNATIVES

Alternatives to the proposed approach are as follows:

2.1. Utilize a variant of the German or Japanese concept with mechanical matrix overcoating of particles prior to compacting and a resin based matrix.

2.2. Utilize FSV process (petroleum pitch matrix), and accept fuel compacts with higher levels of damaged coated fuel particles. With this alternative, some modifications would be required to the reactor design conditions, with a need to accept higher levels of fission product release to the primary coolant circuit.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected approach is to evaluate compacting processes based upon both resin matrix and petroleum pitch matrix and to select a reference process with the other process serving as backup.

The alternative 2.1 approach was not selected because this concept has severe limitations on the volume available for coated fuel particles in the fuel compact, and this limits the power density or the fuel lifetime in the reactor core.

The alternative 2.2 approach was not selected because of the uncertainty in being able to meet the reactor radionuclide requirements without a pressurized secondary containment and active cooling systems, and the risk of delays at the time of construction of a fuel fabrication facility.

4. SCHEDULE REQUIREMENTS

Fuel compacts shall be required for screening or developmental capsule irradiation tests by [TBD]. Fuel compact process conditions and process specifications shall be documented and approved by the ORNL program manager [three] months prior to the start of fuel manufacturing for qualification irradiation tests.

5. PRIORITY

Urgency: TBD
Cost Benefit: H
Uncertainty in existing data: M
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

The fallback position would be the alternative 2.2 approach (i.e., the FSV process). The consequences of nonexecution would be a risk of inadequate performance of qualification fuel, the potential for a requirement to add a pressurized secondary containment to the reactor plant, and potential schedule delays in demonstrating the fuel for a first reactor plant.

7. REFERENCES

1. "MHTGR Fuel Process and Quality Control Description," DOE-HTGR-90257, September 1991.

2. Nabielek, H., et al., "Development of Advanced HTR Fuel Elements," Nuclear Engineering & Design, 1990.

3. Saito, S., et al, "Safety Requirements and Research and Development on HTTR Fuel," Report IWG-GCR/25 International Atomic Energy Agency, 1991

4. Sanders, C.F., and J.D. Sease, “Fabrication and Characterization of Plutonium Test Element FTE-13: An HTGR Test Element Containing PuO_{2-x}, $Th_{0.75}Pu_{0.25}O_{2-x}$ and ThO_2,” ORNL-TM-4207, August 1973.

5. Goodin, D., “MHR-1 Compact Process Development Test Report,” GA Document 910906, Rev. 0, November 1996.

6. "Fuel Product Specification for GT-MHR," DOE-GT-MHR 100209, Rev. 0, May 1994.

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

"AAA" Fuel Program
QUALITY CONTROL TEST TECHNIQUES DEVELOPMENT
DDN RTW.01.07

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The fuel for the DB-MHR must have low levels of as-manufactured defects, as well as properties that ensure that the structural integrity of the coated particles will be maintained during irradiation and postulated accident conditions. The set of fuel product specifications that have been used historically, both in the U.S. and internationally, for coated-particle fuel manufacturing is not sufficiently comprehensive to ensure the acceptable performance of the fuel. There is a need to control additional fuel properties, and new QC methods are needed to measure these properties. Additionally, improved QC methods that can be performed quickly and inexpensively in a glove box environment are needed to support mass production of DF and TF for the DB-MHR.

1.1 Summary of Function/Assumption

"Control Radionuclide Transport from Core," Assumption: Processes are available for manufacturing high quality fuel kernels, coatings, and compacts for inclusion in prismatic fuel elements.

1.2 Current Data Base Summary

The QC techniques currently available for inspection and testing of DB-MHR fuel components are essentially the methods used for inspection of Fort St. Vrain (FSV) fuel (Ref. 1). Many of these methods employ technologies which are inherently time consuming and labor intensive. Although largely adequate for inspection of small quantities of fuel in support of fuel development activities, these techniques require improvement for use in a fuel production facility. Furthermore, many of these methods may not be adaptable to glove box operations for DF and TF fuel manufacturing.

Given that the fission product retentiveness of the SiC coating is of paramount importance to modular gas-cooled reactor design, the best measure of the quality of the as-manufactured SiC coating would be a Cs-release test since this test would directly measure the property of interest. However, because the difficulties and turnaround time associated with a Cs-release test disqualify it for routine QC use, a SiC defect fraction specification has been used historically to control the quality of as-manufactured SiC coatings. The method that has been universally used to measure the defective SiC coating fraction is the burn-leach test. However, the sensitivity of the burn-leach test is limited by gas and liquid transport through pores smaller than about 1 μm; consequently, the test may detect only a subpopulation of the defective particles that contribute to fission product release. In fact, for higher quality fuel, the relative contribution of smaller defects that are not detected by the burn-leach test may increase, assuming such defects are present. Additionally, use of the burn-leach test may be problematic for DB-MHR fuel because of the lower reactivity of DF and TF fuel kernels versus uranium kernels.

The poor performance of the fuel in capsule HRB-21 and in the NPR capsules was due primarily to SiC failure caused by excessive irradiation induced anisotropic shrinkage of IPyC coatings (Ref. 2). Structural isotropy is known to be a key material property correlated with PyC dimensional stability under irradiation, but the optical anisotropy values (BAF_o) measured using the current QC technique do not correlate well with OPyC failure data. Furthermore, the BAF_o values measured for the IPyC coatings in the fuel irradiated in HRB-21 and in the NPR capsules were well within the specified limits.

The IPyC coating must be essentially impermeable at the SiC coating temperature ($\geq$1500 ^{o}C) to prevent HCl gas, a by-product of the thermal decomposition of methyltrichlorosilane, from reaching and reacting with the kernel during SiC coating. Historically, the permeability of the IPyC coating layer has been "controlled" by the IPyC coating density specification, and the effectiveness of the IPyC in protecting the kernel has been "controlled" by the fuel dispersion specification applicable to the fuel particles in fired fuel compacts. The QC method historically used to measure the IPyC coating density is the liquid gradient column (or "sink float") method. This method has been shown to be unreliable if the coating has substantial open porosity, in which case the gradient column fluid fills the open pores giving an erroneously high measured density. Also, the sensitivity of the fuel dispersion measurement to detect low levels of dispersion that could be significant with respect to the formation of SiC defects during SiC coating or during irradiation is suspect.

SiC coatings with excessive free carbon in the grain boundaries, poor grain structure, internal defects such as soot or impurity inclusions or stacking faults, surface cracks, excessive surface roughness, or other unfavorable properties may cause the SiC coating to be prone to fission product leakage or structural failure during irradiation. From a QC standpoint, the SiC defects that are controlled by the defective SiC specification and that are measured by the burn-leach test are defects that penetrate the SiC coating. Latent SiC defects, such as poor microstructure or inclusions within the SiC coating, that would not be detected by the burn-leach test, but that could cause the SiC coating to fail during irradiation, are currently only indirectly "controlled" by the SiC density specification and the requirement that the SiC be β-phase SiC.

The irradiation performance of TRISO-coated PuO_x fuel is known to be sensitive to the O/Pu ratio. Qualified methods are readily available for measuring the O/Pu ratio in bulk samples, but no method is currently available for measuring the O/Pu ratio in individual fuel particles. Furthermore, it is important to measure the O/Pu ratio following deposition of the SiC coating layer.

1.3 Data Needed

The following qualified and documented QC methods are needed for characterizing DF and TF fuel. The QC methods, equipment, and procedures developed for DF and TF fuel must account for the considerable radiological hazards posed by the Pu, Np, Am, and Cm feedstocks.

- An improved method of measuring the fission product retentiveness of the SiC coating; the method could either directly measure fission product retention or be a more sensitive method of measuring SiC defects
- A QC method that provides direct information about PyC microstructure and is sufficiently sensitive to detect small microstructural changes that influence irradiation induced dimensional change
- An improved method of characterizing the impermeability of the IPyC coating
- QC methods capable of detecting SiC defects (such as soot and impurity inclusions, poor grain structure, etc.) that do not penetrate the SiC coating and would therefore not be detected by the burn-leach test, but that could lead to SiC failure during irradiation or accident conditions
- A method to measure the O/Pu ratio in the kernels of individual fuel particles, including after deposition of the various coating layers
- Nondestructive QC methods that can be automated for high throughput, near real time inspection and that can be performed quickly and inexpensively in a glove box environment to support mass production of DF and TF for the DB-MHR

1.4 Data Parameters/Service Conditions

The QC methods must be capable of measuring the product attributes to determine conformance to the fuel product specifications with high confidence. The methods developed for use in mass production of DB-MHR fuel should be nondestructive, whenever practical, and should be capable of being fully automated to provide high-throughput, near real time inspection of fuel product and in-process material.

2. DESIGNER'S ALTERNATIVES

The alternative to the proposed approach is as follows:

2.1. Base the acceptability of DB-MHR fuel on the traditional limited set of product specifications and to rely upon process specifications to control other unspecified properties that are important to the irradiation performance of the fuel. In this approach, the manufacturing processes must be qualified through irradiation testing of fuel manufactured over the full range of acceptable process parameters.

Also, the existing QC methods must be adapted for use in glove boxes.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected approach is to develop the QC methods needed both for improved characterization and to support economical mass production of DB-MHR fuel. Documentation will be prepared for qualifying the new/improved methods and a detailed inspection procedure will be written for each method. Fuel samples for irradiation tests will be inspected using candidate new/improved methods, and the results of the irradiation tests will be used to formally qualify the methods and to establish fuel product specifications based on the new/improved methods.

The alternative approach was not selected because it would involve a higher risk of acceptance of fuel that will not meet DB-MHR fuel performance requirements. Additionally, heavy reliance on process specifications, in lieu of product specifications, would virtually eliminate any flexibility in the fuel manufacturing processes. All processes would have to be frozen, thereby eliminating any ability to implement process improvements that could reduce the cost of fuel manufacturing. Furthermore, there would be no basis for accepting product that was fabricated using process parameters outside of specified limits, which could result in rejection of otherwise acceptable product and a corresponding unnecessary reduction in process yield.

The alternate approach of using the traditional QC methods for mass production of DB-MHR fuel (as opposed to developing improved automated methods) was not selected because fuel unit production costs associated with the use of these methods would be excessively high.

4. SCHEDULE REQUIREMENTS

A complete set of QC methods that can be used in glove boxes, including the new methods that are needed to improve the characterization of the fuel, must be available for use in time to characterize the fuel for the qualification irradiation tests. The improved QC methods needed for mass production of DB-MHR fuel are not necessarily needed to support fuel development and fuel qualification, but must be available for use in the DB-MHR Fuel Fabrication Pilot Plant.

5. PRIORITY

Urgency: TBD
Cost Benefit: H
Uncertainty in existing data: M
Importance of new data: M

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

The fallback position would be the alternate approach discussed in Item 2. The consequences of this approach would be to increase the risk that unacceptable fuel would be delivered to the reactor, with an accompanying higher risk of radionuclide releases from the reactor core. Consequently, there is a risk that this approach would be unacceptable to the customer and/or the NRC. Also, without adequate QC methods to provide objective evidence that as-manufactured fuel has all of the properties essential to acceptable in-service performance, the NRC may require that the DB-MHR include a secondary pressurized containment and active cooling systems.

The fallback position would also eliminate any flexibility to evaluate and implement cost reduction initiatives in the DB-MHR fuel manufacturing processes. This obstacle could preclude ultimate achievement of a competitive fuel cycle cost for the DB-MHR.

7. REFERENCES

1. "MHTGR Fuel Process and Quality Control Description," DOE-HTGR-90257, September 1991.

2. "MHTGR TRISO-P Fuel Failure Evaluation Report," DOE-HTGR-90390, October 1993.

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

“AAA” Fuel Program
PRODUCT RECYCLE AND WASTE TREATMENT TECHNOLOGY
DDN RTW.01.08

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

In order to have a fully qualified fuel manufacturing process, techniques must be developed to recover feedstocks or dispose of contaminated scrap which is generated during each manufacturing step and to convert process waste streams for re-use or disposal. The recovered feedstocks will be recycled into the manufacturing processes.

1.1 Summary of Functions/Assumptions

"Control Transport of Radionuclides from Core," Assumption: Processes are available for manufacturing high quality kernels, coatings and fuel compacts for inclusion in prismatic fuel elements.

1.2 Current Data Base Summary

The recovery of uranium from scrap materials was demonstrated in the US during the manufacture of Fort St. Vrain fuel (33,000 kg heavy metal), when process units were developed to recover highly enriched uranium on a continuous basis (e.g., Ref. 1). Of particular note, the experience applicable to the crushing and dissolving of out-of-specification TRISO-coated particles to recover the heavy metal is extensive, including data from fuel reprocessing developmental tests under a previous DOE program in the 1970s (Ref. 2).

The US nuclear weapons programs and previous civilian nuclear reprocessing programs have developed processes and techniques for recovering transuranics from a wide variety of process streams for recycle and/or disposal. Much of this process technology should be directly applicable to the processes which will be used to fabricate Driver and Transmutation Fuel for the DB-MHR, including the liquid waste streams generated during kernel manufacturing and the gaseous waste streams generated during coating.

Each of the steps for fuel fabrication, from kernels through fuel compacts, will generate substantial quantities of liquid and solid waste products. In addition, a number of the waste streams will contain mixed wastes which need special attention.

1.3 Data Needed

Process flow sheets must be developed, and equipment and operational procedures must be demonstrated for the recovery and recycle of transuranics from reject fuel and from process streams for each step in the manufacture of Driver and Transmutation Fuel for the DB-MHR. Waste minimization shall be a top priority for each process step. The processes, equipment and procedures developed for product recovery and waste treatment must account for the considerable radiological and chemical hazards posed by the Pu/Np/Am/Cm feedstocks and by certain process chemicals (e.g., the methyltrichlorosilane used in SiC coating. All process equipment shall be designed to be critically safe by geometry.

In order to provide adequate accountability for fissile materials, improved software and procedures must be developed to provide real time data on the quantities of transuranics in all process streams. Besides the process systems and equipment, improved software and procedures must be developed to provide real time data on the distribution of SNM in the waste effluents.

All fuel process fabrication operations must include methods to identify mixed waste streams and features to minimize the risks for mixing transuranics with poorly controlled waste materials.

1.4 Data Parameters/Service Conditions

The recovery units must be sized to support process units having minimum batch sizes of [TBD] kg of transuranics in any unit, and for a minimum throughput of [TBD] kg per day. The process yield must be ≥ 90% in any single recovery step, and the total losses of transuranics must be not greater than 0.5% of the input, based on accounting for all quantities in the acceptable product, the in-process losses and the portion recovered from scrap.

2. DESIGNER'S ALTERNATIVES

An alternative to the proposed approach is to use fuel recovery and materials control procedures which are more nearly the same as those developed for production of Fort St. Vrain fuel.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The development approach will include improving the process methods to recover transuranics from scrap fuel components, preparation of process and equipment specifications and flowsheets, and operation of a pilot scale demonstration unit.

The selected approach will include developing process methods to reduce the volumes of waste streams, preparation of process flowsheets and equipment specifications, and operation of a pilot scale waste processing unit.

The alternative was not selected because the Fort St. Vrain fuel manufacturing processes were for uranium and thorium fuels. Moreover, the FSV VSM kernel fabrication process did not produce the large liquid process streams which are characteristic of gelation type processes. Also, at the time of Fort St. Vrain the requirements for rapid evaluations of special nuclear material inventories were less stringent than in current regulations.

4. SCHEDULE REQUIREMENTS

The processes for recovering uranium scrap must be available, and all procedures documented, [three] months prior to the start of fabrication of proof test fuel.

5. PRIORITY

Urgency: TBD
Cost benefit: H
Uncertainty in existing data: M
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

The alternative approach would need to be pursued, with some increase in risk of schedule delays.

7. REFERENCES

1. "MHTGR Fuel Process and Quality Control Description," DOE-HTGR-90257, September 1991.

2. Young, D. T., "Fluidized Combustion of Particles in Reprocessing HTGR Fuel," GA-A14327, March 1977.

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

"AAA" Fuel Program
MINOR ACTINIDE LOSS RATES FROM TF KERNELS DURING PROCESSING
DDN RTW.02.01

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Of the minor actinides prominent in spent LWR fuel, americium is notable in that the partial pressures of Am vapor over the metal are orders of magnitude higher than that for Pu vapor over plutonium and for U vapor over uranium. Even though the proposed fuel is an oxide, or possibly an oxide-carbide mixture, high partial pressures of Am vapor are characteristic of these phases as well. Since both the fabrication of fuel kernels and their subsequent TRISO coating include high-temperature process steps (up to 1600 °C), there is a potential for significant americium volatilization and loss from the TF kernels.

1.1 Summary of Function/Assumptions

"Retain Radionuclides in Fuel Particles with TRISO Coatings," Assumption: Processes are available for manufacturing coated fuel particles that maintain high integrity under irradiation and accident conditions.

1.2 Current Data Base Summary

Cordfunke, et al (Ref. 1), provided a critical review of the Am-O system in 1990. This review supercedes the review of the Am metal system by Hultgren, et al (Ref. 2) in 1973. These data demonstrate, e.g., at 1400°C that the uranium partial pressure over U metal is $\sim 10^{-9.5}$ atm, the plutonium partial pressure over plutonium metal is $\sim 10^{-5.5}$ atm, and the americium partial pressure over americium metal is $\sim 10^{-2.1}$ atm. These trends continue, but at lower partial pressures, over UO_2, Pu_2O_3, and Am_2O_3, which appear to be equally stable. Consequently, americium volatilization and loss have been a concern during the fabrication of fuel elements and targets containing americium in combination with a variety of other actinide oxides.

Experimental evidence of americium volatility was seen during the vacuum casting of metallic IFR fuel that was intended to contain 2.1% Am and 1.3% Np (Ref. 3). Up to 40% of the Am was lost at 1465ºC which is not surprising since the Am vapor pressure of $>10^{-2}$ atm was approximately that of the applied vacuum.

Am-containing targets have been prepared by infiltration of porous $MgAl_2O_4$ pellets with an americium nitrate solution followed by drying, calcining, and finally sintering at 1650 °C under a Ar/H_2 atmosphere (Ref. 4). No Am loss from the target was reported, but the Am reacted with the matrix, apparently forming $AmAlO_3$ and producing a nonuniform distribution of Am within the pellet.

The TRISO-coated Pu fuels irradiated in Peach Bottom FTE-13 and in Dragon and Studsvik were fabricated from highly enriched Pu feedstock (89% and 85% Pu-239, respectively). Somewhat surprisingly, no Am-241 was reported in the elemental analyses of the as-manufactured kernels, and no comments were made regarding Am loss during high-temperature process steps (however, the Belgo Nucleaire researchers who fabricated the $PuO_x/20C$ kernels were concerned about Pu volatilization and loss during processing). It is conceivable that both programs procured "fresh" highly enriched Pu feedstock with a low Am-241 content to avoid the gamma radiation associated with the latter.

The loss of lithium from the TRISO-coated $LiAl_5O_8 \cdot LiAlO_2$ particles during fabrication was a concern for the manufacture of targets for the New Production Reactor (Ref. 5). Li loss rates were reduced to acceptable levels by choosing a more refractory kernel composition and by modifying the standard kernel and coating process conditions. Whether or not there are analogs which will serve to minimize Am loss rates from TF particles remains to be determined.

1.3 Data Needed

The loss rates of americium from TF fuel kernels shall be quantified for every process step during kernel fabrication and coating with emphasis on high-temperature kernel sintering and the initial stages of coating (i.e., through the application of the IPyC coating). While the primary concern is Am volatilization at high temperature, the loss rates of all significant transuranics (i.e., Pu, Np, Cm, etc.) shall also be investigated for each process step in the interest of materials accountability, waste minimization, and trace contamination of the pyrocarbon coatings and the coaters themselves. Precise control of the process atmospheres (Ar, H_2, CO, etc.) and temperatures to minimize americium loss will be emphasized.

If control of process conditions are insufficient to minimize Am losses to acceptable levels, adjustment of the kernel composition and/or introduction of Am_2O_3-containing double oxides, such as Al_2O_3, will be considered.

If the above options fail to reduce Am loss rates from the kernel to acceptable levels, a low-temperature PyC seal coat will be applied directly on the kernel and cleaned (e.g., by gaseous HCl leaching) to remove Am contamination from the seal coat prior to applying the buffer layer.

1.4 Data Parameters/Service Conditions

The standard process conditions for kernel fabrication, TRISO coating, and compacting are defined in a series of process specifications (e.g., Refs. 6, 7 and 8, respectively). These standard process conditions will serve as the point of departure for the developmental studies that will be required to optimize these processes for the production of Transmutation Fuel as called for in DDNs

RTW.01.02, RTW.01.04, and RTW.01.06. The allowable Am loss rate for each process step in kernel fabrication and coating is defined below:

Process Step	Allowable Am Loss (%)	Minim Yield (%)
Kernel Fabrication	[TBD]	[TBD]
Resin Loading	[TBD]	[TBD]
[pH Adjustment]	[TBD]	[TBD]
Drying	[TBD]	[TBD]
Devolatilization	[TBD]	[TBD]
Conversion	[TBD]	[TBD]
[Sintering]	[TBD]	[TBD]
TRISO Coating	[TBD]	[TBD]
[Kernel Seal Coat]	[TBD]	[TBD]
Buffer Layer	[TBD]	[TBD]
IPyC Layer	[TBD]	[TBD]
SiC Layer	[TBD]	[TBD]
OPyC Layer	[TBD]	[TBD]

Sensitivity tests shall be conducted to determine the effect of key process parameters on the product attributes. These tests shall include varying the key process parameters over the range of conditions necessary to establish operating envelopes that support producing material that will meet the fuel specification.

2. DESIGNER'S ALTERNATIVES

Alternatives to the proposed approach are as follows:

2.1. Use standard process conditions. Employ appropriate cleaning techniques to remove elemental Am and/or volatile Am compounds from the accessible surfaces of the finished kernels prior coating (and from any kernel seal coat that may be applied) and recycle the americium.

2.2. Use standard process conditions. Perform an irradiation program to characterize the performance the resulting Transmutation Fuel and design the reactor to accommodate this performance.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected design approach is to development and qualify a particle design and to establish process conditions and equipment that minimizes Am loss rates to acceptable levels.

Alternative 2.1 was not chosen because Am recovery and recycle is expected to be expensive and to increase occupational exposure.

Alternative 2.2 was not chosen because the performance of such fuel is anticipated to be relatively poor, necessitating an uneconomical plant design. Moreover, such Transmutation Fuel may be unsuitable for direct disposal in a geological repository.

4. SCHEDULE REQUIREMENTS

TF particles shall be required for screening or developmental capsule irradiation tests by [TBD]. Process conditions and process specifications shall be established [three] months prior to the start of fuel manufacturing for qualification irradiation tests.

5. PRIORITY

Urgency: TBD
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

The fallback position would be to minimize Am loss rates to the extent practical by kernel design and process optimization and to introduce product cleaning and Am recovery and recycle steps in the process where needed (Alternative 2.1). The consequence of nonexecution would to raise feasibility issues regarding the proposed Transmutation Fuel.

7. REFERENCES

1. Cordfunke, E. H. P., et al., “Thermochemical Data for Reactor Materials and Fission Products,” Elsevier Science Publishers, Amsterdam/New York, ISBN-0-444-88485-8, (1990).

2. Hultgren, R., et al., “Selected Values of the Thermodynamic Properties of the Elements,” American Society for Metals, Metals Park, OH (1973).

3. Trybus, C.L., et al., “Casting of Metallic Fuel Containing Minor Actinide Additions,” J. Nucl. Mat., **204** (1993) 50-55.

4. Konings, R.J.M., et al., “The EFTTRA-T4 Experiment on Americium Transmutation,” J. Nucl. Mat., **282** (2000) 159-170.

5. Thomas, R., et al., “NPR-MHTGR Tritium Target Performance Demonstration and Qualification Program Plan,” EGG-NPRD-8797, EG&G Idaho, Inc., December 1989.

6. "Commercial 450 MW(t) MHTGR Kernel Fabrication Process Specification," DOE-HTGR-90382, September, 1993.

7. "Process Specification for TRISO Coating," DOE-HTGR-90381, August, 1993.

8. "Process Specification for Compact Fabrication," DOE-HTGR-990380, August 1993.

Originator	Date
Engineering Manager	Date
Project Manager	Date

DATE: 09/27/02

“AAA” Fuel Program
DF COATING MATERIAL PROPERTY DATA
DDN RTW.02.02

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Models and codes for predicting stresses in coatings of TRISO coated particles under irradiation and for predicting the probability of failure of each coating layer have been developed for use in design of MHRs. These models require data on the mechanical properties of each coating layer under irradiation. Confirmatory data will be required for the particular PyC and SiC coatings deposited on the DF particles.

1.1 Summary of Functions/Assumptions

"Retain Radionuclides in Fuel Particles," Assumption: Reference fuel failure models are accurate to predict failure under normal operating conditions within a factor of [4] at 95% confidence.

"Retain Radionuclides in Fuel Particles, "Assumption: Reference fuel failure models are sufficiently accurate to predict failure under accident conditions to within a factor of [4] at 95% confidence.

1.2 Current Data Base Summary

Analytical models and codes have been developed to predict the mechanical behavior of coated particles under irradiation require material properties over the range of conditions present in the MHR core (Refs. 1 and 2). Currently, the strength of the SiC layer is established primarily from data on unirradiated SiC, and there is limited information on the effects of fast fluence; a lower than measured strength is assumed to account for fast fluence effects. The Weibull modulus used in the codes is for unirradiated SiC. Material property correlations for pyrocarbons under irradiation have been developed, but the uncertainties are large.

1.3 Data Needed

Additional material property data are required for isotropic pyrocarbon coatings, and high-density, β-phase silicon carbide coatings as applied to Driver Fuel particles in compacts.

Data are needed to establish correlations between the following key material properties and behaviors: buffer porosity and available void volume and its change during irradiation; PyC dimensional change, and isotropy (Bacon

Anisotropy Factor or corresponding Optical Anisotropy Factor) versus PyC failure in-reactor; SiC strength, Weibull modulus, and microstructure versus resistance to diffusion of fission products and mechanical failure during irradiation. Quality assurance must be in accordance with the requirements for experimental data which are "safety-related."

1.4 Parameters/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications as appropriate:

Parameter	Value
Reference fuel	TRISO-coated DF in compacts
Fuel burnup range	[0 to 85]% FIMA
Peak fast fluence	8×10^{25} n/m^2 (E >29 fJ)
System pressure[*]	
Normal condition	7.1 MPa (70 atm)
Off-normal conditions	0.1 to 7.1 MPa (1 to 70 atm)
Fuel temperatures/time	
Normal condition	[800 to 1400 C]/≤36 months
Off-normal conditions (core conduction cooldown)	
Pressurized (wet)	[≤1400 °C]/≤100 h
Depressurized	[1200 to 2000 °C] /100 to 1000h
Environment	
Normal conditions	He/Ne(in-pile tests)
Off-normal conditions	He; He/H_2O/CO/H_2; He/CO/N_2
Coolant/Impurity Levels	
Normal condition (Design):	Total oxidants [≤7Pa (1ppmv)]
H_2O	[0.07 to 0.7 Pa (0.01 to 0.1 ppmv)]
CO	[1.5 to 6 Pa (0.2 to 8 ppmv)]
CO_2	[0.2 to 1.5 Pa 0.03 to 0.2 ppmv)]
H_2	[3 to 10 Pa (0.5 to 1.5 ppmv)]
CH_4	[0.3 to1.5 Pa (0.5 to 1.5 ppmv)]
N_2	[3 Pa (0.4 ppmv)]
Pressurized condition	
H_2O	[0 to TBD KPa (0 to TBD atm)]
Depressurized condition	
CO	[0 to 35 KPa (0 to 0.35 atm)]
N_2	[0 to 65 KPa (0 to 0.65 atm)]
Expected failure fraction	
Normal conditions	[10^{-5} to 10^{-3}]
Off-normal conditions	[10^{-4} to 10^{-2}]

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[*] A lower pressure may be used to satisfy DDN if adequate technical justification is provided.

2. DESIGNER'S ALTERNATIVES

Alternatives to the acquisition of the above data are:

2.1 Consider that fuel particle design is fixed and use kernel and coating properties which correspond to attributes of previously tested fuel (i.e., rely upon the existing data base with its attendant uncertainties).

3. SELECTED DESIGN APPROACH AND EXPLANATION

The development effort will include preparing, irradiating and evaluating special test samples and coated particles to obtain data needed for design. Alternative 2.1 would likely result in fuel performance uncertainties in excess of the 4x predictive accuracy goal adopted for Driver Fuel.

4. SCHEDULE REQUIREMENTS

Preliminary data [6] months before start of Preliminary Design and final data [12] months prior to completion of Final Design.

5. PRIORITY

Urgency: TBD
Cost benefit: TBD
Uncertainty in existing data: M
Importance of new data: H

6. CONSEQUENCES ON NONEXECUTION AND FALLBACK POSITION

The fallback position is to use a qualified particle design made with established process parameters and equipment but with higher predicted failure rates. This position will increase the probability that DF performance requirements will not be satisfied.

7. REFERENCES

1. Myers, B.F., “Fuel Design Data Manual,” GA Document 901866, Ref. F, General Atomics, August 1987.

2. Ho., F., “Material Models of Pyrocarbon and Pyrolytic Silicon Carbide,” GA Document CEGA-002820, Rev. 1, July 1993.

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

"AAA" Fuel Program
TF COATING MATERIAL PROPERTY DATA
DDN RTW.02.03

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Models and codes for predicting stresses in coatings of TRISO coated particles under irradiation and for predicting the probability of failure of each coating layer have been developed for use in the design of MHRs. These models require data on the mechanical properties of each coating layer under irradiation. Confirmatory data will be required for the particular PyC and SiC coatings deposited on the Transmutation Fuel (TF) particles; these data will be particularly important if nonstandard coating conditions are required for the TF particles to minimize Am losses during fabrication.

1.1 Summary of Functions/Assumptions

"Retain Radionuclides in Fuel Particles," Assumption: Reference fuel failure models are accurate to predict failure under normal operating conditions within a factor of [10] at 95% confidence.

"Retain Radionuclides in Fuel Particles, "Assumption: Reference fuel failure models are sufficiently accurate to predict failure under accident conditions to within a factor of [10] at 95% confidence.

1.2 Current Data Base Summary

Analytical models and codes to predict the mechanical behavior of coated particles under irradiation require material properties over the range of conditions present in the MHR core (Refs. 1 and 2). Currently, the strength of the SiC layer is established primarily from data on unirradiated SiC, and there is limited information on the effects of fast fluence; a lower than measured strength is assumed to account for fast fluence effects. The Weibull modulus used in the codes is for unirradiated SiC. Material property correlations for pyrocarbons under irradiation have been developed, but the uncertainties are large.

1.3 Data Needed

Additional material property data are required for the isotropic pyrocarbon coatings, and high-density, β-phase silicon carbide coatings as applied to TF particles in compacts.

Data are needed to establish correlations between the following key material properties and behaviors: buffer porosity and available void volume and its change during irradiation; PyC dimensional change and isotropy (Bacon Anisotropy Factor or corresponding Optical Anisotropy Factor) and versus PyC failure in-reactor; SiC strength, Weibull modulus, and microstructure versus resistance to diffusion of fission products and mechanical failure during irradiation. Quality assurance must be in accordance with the requirements for experimental data which are "safety-related."

1.4 Parameters/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications as appropriate:

Parameter	Value
Reference fuel	TRISO-coated TF particles in compacts
Fuel burnup range	[0 to 85]% FIMA
Peak fast fluence	8×10^{25} n/m^2 (E >29 fJ)
System pressure[*]	
Normal condition	7.1 MPa (70 atm)
Off-normal conditions	0.1 to 7.1 MPa (1 to 70 atm)
Fuel temperatures/time	
Normal condition	[800 to 1400 C]/≤36 months
Off-normal conditions (core conduction cooldown)	
Pressurized (wet)	[≤1400 °C]/≤100 h
Depressurized	[1200 to 2000 °C] /100 to 1000h
Environment	
Normal conditions	He/Ne(in-pile tests)
Off-normal conditions	He; He/H_2O/CO/H_2; He/CO/N_2
Coolant/Impurity Levels	
Normal condition (Design):	Total oxidants [≤7Pa (1ppmv)]
H_2O	[0.07 to 0.7 Pa (0.01 to 0.1 ppmv)]
CO	[1.5 to 6 Pa (0.2 to 8 ppmv)]
CO_2	[0.2 to 1.5 Pa 0.03 to 0.2 ppmv)]
H_2	[3 to 10 Pa (0.5 to 1.5 ppmv)]
CH_4	[0.3 to1.5 Pa (0.5 to 1.5 ppmv)]
N_2	[3 Pa (0.4 ppmv)]
Pressurized condition	[0 to TBD KPa (0 to TBD atm)]
H_2O	
Depressurized condition	[0 to 35 KPa (0 to 0.35 atm)]
CO	[0 to 65 KPa (0 to 0.65 atm)]
N_2	
Expected failure fraction	
Normal conditions	[10^{-5} to 10^{-3}]
Off-normal conditions	[10^{-4} to 10^{-2}]

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[*] A lower pressure may be used to satisfy DDN if adequate technical justification is provided.

2. DESIGNER'S ALTERNATIVES

Alternatives to the acquisition of the above data are:

2.1 Consider that fuel particle design is fixed and use kernel and coating properties which correspond to attributes of previously tested fuel (i.e., rely upon the existing data base with its attendant uncertainties).

3. SELECTED DESIGN APPROACH AND EXPLANATION

The development effort will include preparing, irradiating and evaluating special test samples and coated particles to obtain data needed for design. Alternative 2.1 would likely result in fuel performance uncertainties in excess of the 10x predictive accuracy goal adopted for TF fuel performance predictions.

4. SCHEDULE REQUIREMENTS

Preliminary data [6] months before start of Preliminary Design and final data [12] months prior to completion of Final Design.

5. PRIORITY

Urgency: TBD
Cost benefit: TBD
Uncertainty in existing data: H
Importance of new data: H

6. CONSEQUENCES ON NONEXECUTION AND FALLBACK POSITION

The fallback position is to use a qualified particle design made with established process parameters and equipment but with higher predicted failure rates. This position will increase the probability that TF performance requirements will not be satisfied.

7. REFERENCES

1. Myers, B.F., “Fuel Design Data Manual,” GA Document 901866, Ref. F, General Atomics, August 1987.

2. Ho., F., “Material Models of Pyrocarbon and Pyrolytic Silicon Carbide,” GA Document CEGA-002820, Rev. 1, July 1993.

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

“AAA” Fuel Program
GAS PRODUCTION IN DF PARTICLES
DDN RTW.02.04

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Gaseous species are produced in the DF particles (95% Pu/5% Np) during irradiation by three primary mechanisms: (1) Kr and Xe isotopes from fission, (2) He from alpha particles resulting from certain nuclear reactions and from decay of certain radionuclides, and (3) CO from the reaction of excess oxygen liberated during fission with the carbonaceous buffer layer. All three types of gases contribute to kernel swelling and to internal pressure buildup. In addition, CO apparently activates kernel migration and may react with and degrade the SiC coating at high temperature.

1.1 Summary of Function/Assumptions

"Retain Radionuclides in Fuel Particles," Assumption: Reference fuel failure models are sufficiently accurate to within a factor of [4] at 95% confidence.

"Protect the Capability to Retain Radionuclides in Fuel Particles," Assumption: Reference fuel failure models are sufficiently accurate to predict failure during core heatup transients to within a factor of [TBD] at 95% confidence.

"Control Transport from Core," Assumption: Adequate data is available to predict fuel performance under transient conditions.

"Retain Radionuclides in Fuel Particles," Assumption: Reference fuel failure models are sufficiently accurate to predict failure under transient conditions to within a factor of [4x] at 95% confidence.

1.2 Current Data Base Summary

Nuclear analysis methods are available to predict the isotopic composition of Driver Fuel as a function of burnup under DB-MHR core conditions with reasonable confidence. (However, there are significant uncertainties regarding the fractions of the Kr and Xe isotopes that are released from the DF fuel kernel as a function of burnup, temperature and time; this aspect is addressed in DDN.RTW.03.01).

The production of CO in oxidic fuel kernels is a complex thermochemical process which has been investigated both theoretically and experimentally for a number of different kernel compositions (e.g., Refs. 1 through 4). CO is generated in oxide fuel kernels as a function of the initial actinide content, O/Me ratio, burnup and temperature because the fission product atoms are not capable of reacting with all of the oxygen liberated when a heavy-metal atom is fissioned; this "excess" oxygen may then diffuse out of the kernel and react with the carbonaceous buffer layer to form CO. The generation of CO is problematic: (1) it contributes to kernel swelling and the internal gas pressure, (2) it apparently activates kernel migration, and (3) it may react with and degrade the SiC coating at high temperature.

Two practical methods have been devised to suppress CO formation in oxidic fuel particles. With uranium fuels, a mixture of UC_2 and UO_2 ("UCO") has successfully been used (e.g., Ref. 1); the excess oxygen reacts preferentially with the UC_2 rather than the buffer carbon. The optimum amount of UC_2 in the kernel is determined by the peak burnups with higher burnups requiring greater amounts of the carbide. With plutonium fuels, a substiochiometric oxide kernel has been demonstrated to be effective: particles with a O/Pu ratio = 1.68 showed insignificant kernel migration when irradiated to 70% FIMA at ~1440 ^{o}C in FTE-13 whereas other particles in the same test with a O/Pu ratio = 1.8 migrated excessively (Ref. 5).

There are no experimental or theoretical data for CO generated during irradiation of Pu or Pu-Np oxide fuel. Whether or not there is sufficient thermochemical data to predict CO generation as a function of burnup remains to be determined. However, thermodynamic predictions agreed reasonably with experimental measurements in ThO_2 and $(Th,U)O_2$ particles irradiated to 17 and 23.1 % FIMA, respectively (Ref. 4).

1.3 Data Needed

The production of gaseous species (Kr, Xe, He, CO) in Driver Fuel particles needs to be determined as a function of initial O/Me ratio, burnup and temperature. The first step will be a theoretical prediction with the best available nuclear and thermochemical analysis methods, but some experimental confirmation, especially with to regard to CO production, will be necessary.

The quantity and speciation of gases produced in reference DF particles shall be determined as a function of burnup and irradiation temperature. Experimental equipment and protocols for this purpose have been successfully demonstrated (e.g., Ref. 6). Samples with a range of O/Me ratios and, possibly, $PuO_{1.5}$–$PuC_{1.5}$ shall be irradiated to full burnup over a range of temperatures, and a postirradiation examination shall be conducted, including the measurement of the gas inventories and the extent of kernel migration.

1.4 Parameters/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications as appropriate:

Parameter	Value
Reference fuel	TRISO-coated DF in compacts
Fuel burnup range	[0 to 85]% FIMA
Peak fast fluence	8 x 10^{25} n/m^2 (E >29 fJ)
System pressure*	
Normal condition	7.1 MPa (70 atm)
Off-normal conditions	0.1 to 7.1 MPa (1 to 70 atm)
Fuel temperatures/time	
Normal condition	[800 to 1400 C]/≤36 months
Off-normal conditions (core conduction cooldown)	
Pressurized (wet)	[≤1400 ^{o}C]/≤100 h
Depressurized	[1200 to 2000 ^{o}C] /100 to 1000h
Thermal Gradient	
Normal conditions	[50 - 200 C^{o}/cm]
Off-normal conditions	[TBD]
Environment	
Normal conditions	He/Ne(in-pile tests)
Off-normal conditions	He; He/H_2O/CO/H_2; He/CO/N_2
Coolant/Impurity Levels	
Normal condition (Design):	Total oxidants [≤7Pa (1ppmv)]
H_2O	[0.07 to 0.7 Pa (0.01 to 0.1 ppmv)]
CO	[1.5 to 6 Pa (0.2 to 8 ppmv)]
CO_2	[0.2 to 1.5 Pa 0.03 to 0.2 ppmv)]
H_2	[3 to 10 Pa (0.5 to 1.5 ppmv)]
CH_4	[0.3 to1.5 Pa (0.5 to 1.5 ppmv)]
N_2	[3 Pa (0.4 ppmv)]
Pressurized condition	
H_2O	[0 to TBD KPa (0 to TBD atm)]
Depressurized condition	
CO	[0 to 35 KPa (0 to 0.35 atm)]
N_2	[0 to 65 KPa (0 to 0.65 atm)]
Expected failure fraction	
Normal conditions	[10^{-5} to 10^{-3}]
Off-normal conditions	[10^{-4} to 10^{-2}]

2. DESIGNER'S ALTERNATIVES

Alternatives to the acquisition of the above described data are:

2.1 Conduct a strictly empirical evaluation program with a variety of kernel compositions, including a range of 1.5 ≤ O/Me ≤ 1.75, and choose a reference DF design on the basis of the irradiation and postirradiation heating results.

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*A lower pressure may be used to satisfy DDN if adequate technical justification is provided.

2.2 Rely upon theoretical analysis with the current nuclear and thermochemical methods to define the reference DF particle. Characterize that DF particle experimentally and design the plant to accommodate the observed performance.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected approach is to measure the production rates of gaseous species, especially CO, in DF particles as a function of initial O/Me ratio, burnup and temperature. Some quantitative understanding of the kernel chemistry of the DF particle should facilitate the definition of an optimal DF kernel design and the definition of an efficient cost-effective characterization program.

Alternative 2.1 is technically viable, but it could require a very extensive and expensive test matrix. Because of the large uncertainties in the current thermochemical methods when applied to the DF particle, Alternative 2.2 would likely require excessive conservatism in the plant design.

4. SCHEDULE REQUIREMENTS

Preliminary data [6] months before start of Preliminary Design and final data [12] months prior to completion of Final Design.

5. PRIORITY

Urgency: TBD
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H

6. CONSEQUENCES OF NONEXECUTION AND FALLBACK POSITION

Alternative 2.1 along with a more conservative fuel and core design to account for the large uncertainties in the current thermochemical performance models. Failure to fully exploit the high-temperature performance capability of TRISO-coated DF particles will necessitate a more conservative fuel and core design which could include unnecessarily restrictive limits on fuel temperatures during normal operation and core conduction cooldown transients.

7. REFERENCES

1. Homan, F.J., et al., “Stoichiometric Effects on Performance of High-Temperature Gas-Cooled Reactor Fuels from the U-C-O System,” Nucl. Tech., **35**, (1977) 428.

2. Proksch, E., et al., “Production of CO during Burnup of UO_2 Kerneled HTR Fuel Particles,” J. Nucl. Mat., **107**, (1982) 280-285.

3. Proksch, E., et al., “Production of CO during Burnup of $(Th,U)O_2$ Kerneled HTR Fuel Particles,” J. Nucl. Mat., **136**, (1985) 129-135.

4. Lindemer, T.B., “Measurement and Interpretation of CO and Kr plus Xe in Irradiated ThO_2-Containing HTGR Particles,” J. Am. Ceram. Soc. 60 (1977) 409–416.

5. Miller, C. M., and W. J. Scheffel, "Postirradiation Examination and Evaluation of Peach Bottom FTE-13," GA Document 906939, Rev. N/C, GA Technologies, November 1, 1985.

6. Lindemer, T.B., and R.L. Pearson, “Apparatus for Measuring Gases from Irradiated Fuel Particles,” J. Nucl. Mat., **55**, 359 (1975).

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

“AAA” Fuel Program
GAS PRODUCTION IN TF PARTICLES
DDN RTW.02.05

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Gaseous species are produced in the TF particles during irradiation by three primary mechanisms: (1) Kr and Xe isotopes from fission, (2) He from alpha particles resulting from certain nuclear reactions and from decay of certain radionuclides, and (3) CO from the reaction of excess oxygen liberated during fission with the carbonaceous buffer layer. All three types of gases contribute to kernel swelling and to internal pressure buildup. In addition, CO apparently activates kernel migration and may react with and degrade the SiC coating at high temperature.

1.1 Summary of Function/Assumptions

"Retain Radionuclides in Fuel Particles," Assumption: Reference fuel failure models are sufficiently accurate to within a factor of [10] at 95% confidence.

"Retain Radionuclides in Fuel Particles," Assumption: Reference fuel failure models are sufficiently accurate to predict failure under transient conditions to within a factor of [10] at 95% confidence.

1.2 Current Data Base Summary

Nuclear analysis methods are available to predict the isotopic composition of Transmutation Fuel as a function of burnup under DB-MHR core conditions with reasonable confidence. (However, there are significant uncertainties regarding the fractions of the Kr and Xe isotopes that are released from the TF fuel kernel as a function of burnup, temperature and time; this aspect is addressed in DDN.RTW.03.01).

The production of CO in oxidic fuel kernels is a complex thermochemical process which has been investigated both theoretically and experimentally for a number of different kernel compositions (e.g., Refs. 1 through 4). CO is generated in oxide fuel kernels as a function of the initial actinide content, O/Me ratio, burnup and temperature because the fission product atoms are not capable of reacting with all of the oxygen liberated when a heavy-metal atom is fissioned; this “excess” oxygen may then diffuse out of the kernel and react with the carbonaceous buffer layer forming CO. The generation of CO is problematic: (1) it contributes to kernel swelling and the internal gas pressure, (2) it apparently activates kernel migration, and (3) it may react with and degrade the SiC coating at high temperature.

Two practical methods have been devised to suppress CO formation in oxidic fuel particles. With uranium fuels, a mixture of UC_2 and UO_2 ("UCO") has successfully been used (e.g., Ref. 1); the excess oxygen reacts preferentially with the UC_2 rather than the buffer carbon. The optimum amount of UC_2 in the kernel is determined by the peak burnups with higher burnups requiring greater amounts of the carbide. With plutonium fuels, a substiochiometric oxide kernel has been demonstrated to be effective: particles with a O/Pu ratio = 1.68 showed insignificant kernel migration when irradiated to 70% FIMA at ~1440 ^{o}C in FTE-13 whereas other particles in the same test with a O/Pu ratio = 1.8 migrated excessively (Ref. 5).

There are no experimental or theoretical data for CO generated during irradiation of TF oxide fuel. Whether or not there is sufficient thermochemical data to predict CO generation as a function of burnup remains to be determined. However, thermodynamic predictions agreed reasonably with experimental measurements in ThO_2 and $(Th,U)O_2$ particles irradiated to 17 and 23.1 % FIMA, respectively (Ref. 4).

1.3 Data Needed

The production of gaseous species (Kr, Xe, He, CO) in Transmutation Fuel particles needs to be determined as a function of initial actinide content, O/Me ratio, burnup and temperature. The first step will be a theoretical prediction with the best available nuclear and thermochemical analysis methods, but some experimental confirmation, especially with to regard to CO production, will be necessary.

The quantity and speciation of gases produced in reference TF particles shall be determined as a function of kernel composition, burnup and irradiation temperature. Experimental equipment and protocols for this purpose have been successfully demonstrated (e.g., Ref. 6). Samples with a range of O/Metal ratios shall be irradiated to full burnup over a range of temperatures, and a postirradiation examination shall be conducted, including the measurement of the gas inventories and the extent of kernel migration.

1.4 Parameters/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications as appropriate:

Parameter	Value
Reference fuel	TRISO-coated TF in compacts
Fuel burnup range	[0 to 85]% FIMA
Peak fast fluence	8×10^{25} n/m^2 (E >29 fJ)
System pressure*	
Normal condition	7.1 MPa (70 atm)
Off-normal conditions	0.1 to 7.1 MPa (1 to 70 atm)

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*A lower pressure may be used to satisfy DDN if adequate technical justification is provided.

Fuel temperatures/time	
Normal condition	[800 to 1400 C]/≤36 months
Off-normal conditions (core conduction cooldown)	
Pressurized (wet)	[≤1400 ^{o}C]/≤100 h
Depressurized	[1200 to 2000 ^{o}C] /100 to 1000h
Thermal Gradient	
Normal conditions	[50 - 200 C^{o}/cm]
Off-normal conditions	[TBD]
Environment	
Normal conditions	He/Ne(in-pile tests)
Off-normal conditions	He; He/H_2O/CO/H_2; He/CO/N_2
Coolant/Impurity Levels	
Normal condition (Design):	Total oxidants [≤7Pa (1ppmv)]
H_2O	[0.07 to 0.7 Pa (0.01 to 0.1 ppmv)]
CO	[1.5 to 6 Pa (0.2 to 8 ppmv)]
CO_2	[0.2 to 1.5 Pa 0.03 to 0.2 ppmv)]
H_2	[3 to 10 Pa (0.5 to 1.5 ppmv)]
CH_4	[0.3 to1.5 Pa (0.5 to 1.5 ppmv)]
N_2	[3 Pa (0.4 ppmv)]
Pressurized condition	
H_2O	[0 to TBD KPa (0 to TBD atm)]
Depressurized condition	
CO	[0 to 35 KPa (0 to 0.35 atm)]
N_2	[0 to 65 KPa (0 to 0.65 atm)]
Expected failure fraction	
Normal conditions	[10^{-5} to 10^{-3}]
Off-normal conditions	[10^{-4} to 10^{-2}]

2. DESIGNER'S ALTERNATIVES

Alternatives to the acquisition of the above described data are:

2.1 Conduct a strictly empirical evaluation program with a variety of kernel compositions, including a range of O/Me ratios, and choose a reference TF design on the basis of the irradiation and postirradiation heating results. It appears from thermochemical calculations that oxycarbide fuel is excluded because Am loss would be too high during carbothermic conversion of the oxide.

2.2 Rely upon theoretical analysis with the current nuclear and thermochemical methods to define the reference TF particle. Characterize that TF particle experimentally and design the plant to accommodate the observed performance.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected approach is to measure the production rates of gaseous species, especially CO, in TF particles as a function of initial O/Me ratio, burnup and temperature. Some quantitative understanding of the kernel chemistry of the DF particle should facilitate the definition of an optimal TF kernel design and the definition of an efficient cost-effective characterization program.

Alternative 2.1 is technically viable, but it could require a very extensive and expensive test matrix. Because of the very large uncertainties in the current thermochemical methods when applied to the TF particle, Alternative 2.2 would likely require excessive conservatism in the plant design.

4. SCHEDULE REQUIREMENTS

Preliminary data [6] months before start of Preliminary Design and final data [12] months prior to completion of Final Design.

5. PRIORITY

Urgency: TBD
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H

6. CONSEQUENCES OF NONEXECUTION AND FALLBACK POSITION

Alternative 2.1 along with a more conservative fuel and core design to account for the large uncertainties in the current thermochemical performance models. Failure to fully exploit the high-temperature performance capability of TRISO-coated TF particles will necessitate a more conservative fuel and core design which could include unnecessarily restrictive limits on fuel temperatures during normal operation and core conduction cooldown transients.

7. REFERENCES

1. Homan, F.J., et al., “Stoichiometric Effects on Performance of High-Temperature Gas-Cooled Reactor Fuels from the U-C-O System,” Nucl. Tech., **35**, 428 (1977).

2. Proksch, E., et al., “Production of CO during Burnup of UO_2 Kerneled HTR Fuel Particles,” J. Nucl. Mat., **107**, (1982) 280-285.

3. Proksch, E., et al., “Production of CO during Burnup of $(Th,U)O_2$ Kerneled HTR Fuel Particles,” J. Nucl. Mat., **136**, (1985) 129-135.

4. Lindemer, T.B., “Measurement and Interpretation of CO and Kr plus Xe in Irradiated ThO2-Containing HTGR Particles,” J. Am. Ceram. Soc. 60 (1977) 409–416.

5. Miller, C. M., and W. J. Scheffel, "Postirradiation Examination and Evaluation of Peach Bottom FTE-13," GA Document 906939, Rev. N/C, GA Technologies, November 1, 1985.

6. Lindemer, T.B., and R.L. Pearson, “Apparatus for Measuring Gases from Irradiated Fuel Particles,” J. Nucl. Mat., **55**, 359 (1975).

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

“AAA” Fuel Program
DF DEFECTIVE PARTICLE PERFORMANCE DATA
DDN RTW.02.06

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Failure of defective particles (particles with as-manufactured defects) is predicted to be a major contributor to fission product release from the cores of modular HTGRs during normal operation and postulated accidents. Single-effects data on the performance of defective DF particles are needed to refine current fuel performance models.

1.1 Summary of Function/Assumptions

"Retain Radionuclides in Fuel Particles," Assumption: Reference fuel failure models are sufficiently accurate to within a factor of [4] at 95% confidence.

"Protect the Capability to Retain Radionuclides in Fuel Particles," Assumption: Reference fuel failure models are sufficiently accurate to predict failure during core heatup transients to within a factor of [TBD] at 95% confidence.

"Retain Radionuclides in Fuel Particles," Assumption: Reference fuel failure models are sufficiently accurate to predict failure under transient conditions to within a factor of [4x] at 95% confidence.

1.2 Current Data Base Summary

Indirect performance data exist for TRISO-coated fuel particles with variable levels of assorted defects. These indirect data along with fundamental material properties have been used to derive analytical models which predict failure of particles with one or more defective layers (Refs. 1 and 2). These defective particles are predicted to be the primary source of failed fuel during normal operation and postulated accidents.

The HRB-21 capsule contained piggy-back samples of LEU UCO particles with missing buffer layers; many of these particles survived irradiation, but these samples have not been completely characterized (Ref. 3). Previously, irradiated TRISO-coated UC_2 particles have been heated with and without the OPyC layer in place, and their high-temperature performance appeared to be largely unaffected by the removal of the OPyC layer.

1.3 Data Needed

Existing data from defective uranium fuel particles provide indirect evidence for how defective DF particles may perform. In order to reduce prediction uncertainty, test data on the performance of irradiated defective DF particles are needed. Data should provide a technically sound basis for deriving performance models for defective particles so that the observed failure at 95% confidence is within [4x] of failure predicted at 50% confidence.

1.4 Parameters/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications as appropriate:

Data are required, as noted above, which will be applicable to the following types of defective particles and range of test/analytical conditions:

Parameter	Value
Reference fuel	TRISO-coated DF in compacts
Defects of interest	Missing buffer HM dispersion in the buffer layer (missing IPyC) Missing or defective SiC layer; intact OPyC layer Missing or failed OPyC layer
Fuel burnup range	[0 to 85]% FIMA
Peak fast fluence	8×10^{25} n/m^2 (E >29 fJ)
System pressure[*]	
Normal condition	7.1 MPa (70 atm)
Off-normal conditions	0.1 to 7.1 MPa (1 to 70 atm)
Fuel temperatures/time	
Normal condition	[800 to 1400 C]/≤36 months
Off-normal conditions (core conduction cooldown)	
Pressurized (wet)	[≤1400 °C]/≤100 h
Depressurized	[1200 to 2000 °C] /100 to 1000h
Thermal Gradient	
Normal conditions	[50 - 200 C°/cm]
Off-normal conditions	[TBD]
Environment	
Normal conditions	He/Ne(in-pile tests)
Off-normal conditions	He; He/H_2O/CO/H_2; He/CO/N_2
Coolant/Impurity Levels	
Normal condition (Design):	Total oxidants [≤7Pa (1ppmv)]
H_2O	[0.07 to 0.7 Pa (0.01 to 0.1 ppmv)]
CO	[1.5 to 6 Pa (0.2 to 8 ppmv)]
CO_2	[0.2 to 1.5 Pa 0.03 to 0.2 ppmv)]
H_2	[3 to 10 Pa (0.5 to 1.5 ppmv)]
CH_4	[0.3 to1.5 Pa (0.5 to 1.5 ppmv)]
N_2	[3 Pa (0.4 ppmv)]

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[*] A lower pressure may be used to satisfy DDN if adequate technical justification is provided.

Pressurized condition	[0 to TBD KPa (0 to TBD atm)]
H_2O	
Depressurized condition	[0 to 35 KPa (0 to 0.35 atm)]
CO	[0 to 65 KPa (0 to 0.65 atm)]
N_2	
Expected failure fraction	
Normal conditions	[10^{-5} to 10^{-3}]
Off-normal conditions	[10^{-4} to 10^{-2}]

2. DESIGNER'S ALTERNATIVES

Alternatives to the acquisition of the above described data are:

2.1 Use the current defective-particle performance models derived from fundamental materials properties and increase design margins to compensate for the large model uncertainties.

2.2 Assume all defective particles fail immediately and design for the predicted higher failure rates; limit the allowable levels of coating defects as necessary to meet fuel failure criteria with the assumption of 100% failure.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Acquire gaseous and metallic fission product release data from TRISO-coated DF particles with missing or defective layers to determine their failure characteristics as a function of environmental conditions, and update the defective particle performance models.

Alternative 2.1 would result in predictive uncertainties in excess of the [4x] predictive accuracy goal. Alternative 2.2 would result in too much conservatism in performance predictions with costly fuel product specification or reactor design changes needed to achieve the required degree of fission product control.

4. SCHEDULE REQUIREMENTS

Preliminary data [6] months before start of Preliminary Design and final data [12] months prior to completion of Final Design.

5. PRIORITY

Urgency: TBD
Cost benefit: M
Uncertainty in existing data: H
Importance of new data: M

6. CONSEQUENCES OF NONEXECUTION AND FALLBACK POSITION

The fallback position is to assume defective DF particles fail immediately. The consequence is expected to be a significantly higher failure fraction prediction than would actually be observed for a given level of as-manufactured defects. Unnecessary conservatism in the performance models will result in unnecessarily stringent fuel product specifications or possible reactor design changes to accommodate unrealistically high, predicted radionuclide levels in the primary circuit.

7. REFERENCES

1. Myers, B.F., "Fuel Design Data Manual," GA Document 901866, Ref. F, General Atomics, August 1987.

2. Ho., F., "Material Models of Pyrocarbon and Pyrolytic Silicon Carbide," GA Document CEGA-002820, Rev. 1, July 1993.

3. Baldwin, C.A., et al., "Fuel Capsule HRB-21 Postirradiation Examination Report," DOE-HTGR-100229, ORNL, April 1995.

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

“AAA” Fuel Program
TF DEFECTIVE PARTICLE PERFORMANCE DATA
DDN RTW.02.07

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Failure of defective particles (particles with as-manufactured defects) is predicted to be a major contributor to fission product release from modular HTGR cores during normal operation and postulated accidents. Single-effects data on the performance of defective TF particles are needed to refine current fuel performance models.

1.1 Summary of Function/Assumptions

"Retain Radionuclides in Fuel Particles," Assumption: Reference fuel failure models are sufficiently accurate to within a factor of [10x] at 95% confidence.

"Protect the Capability to Retain Radionuclides in Fuel Particles," Assumption: Reference fuel failure models are sufficiently accurate to predict failure during core heatup transients to within a factor of [TBD] at 95% confidence.

"Retain Radionuclides in Fuel Particles," Assumption: Reference fuel failure models are sufficiently accurate to predict failure under transient conditions to within a factor of [10x] at 95% confidence.

1.2 Current Data Base Summary

Indirect performance data exist for TRISO-coated fuel particles with variable levels of assorted defects. These indirect data along with fundamental material properties have been used to derive analytical models which predict failure of particles with one or more defective layers (Refs. 1 and 2). These defective particles are predicted to be the primary source of failed fuel during normal operation and postulated accidents.

The HRB-21 capsule contained piggy-back samples of LEU UCO particles with missing buffer layers; many of these particles survived irradiation, but these samples have not been completely characterized (Ref. 3). Previously, irradiated TRISO-coated UC_2 particles have been heated with and without the OPyC layer in place, and their high-temperature performance appeared to be largely unaffected by the removal of the OPyC layer.

1.3 Data Needed

Existing data from defective uranium fuel particles provide indirect evidence for how defective DF particles may perform. In order to reduce prediction uncertainty, test data on the performance of irradiated defective TF particles are needed. Data should provide a technically sound basis for deriving performance models for defective particles so that the observed failure at 95% confidence is within [10x] of failure predicted at 50% confidence.

1.4 Parameters/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications as appropriate:

Data are required, as noted above, which will be applicable to the following types of defective particles and range of test/analytical conditions:

Parameter	Value
Reference fuel	TRISO-coated TF in compacts
Defects of interest	Missing buffer HM dispersion in the buffer layer (missing IPyC) Missing or defective SiC layer; intact OPyC layer Missing or failed OPyC layer
Fuel burnup range	[0 to 85]% FIMA
Peak fast fluence	8×10^{25} n/m^2 (E >29 fJ)
System pressure[*]	
Normal condition	7.1 MPa (70 atm)
Off-normal conditions	0.1 to 7.1 MPa (1 to 70 atm)
Fuel temperatures/time	
Normal condition	[800 to 1400 C]/≤36 months
Off-normal conditions (core conduction cooldown)	
Pressurized (wet)	[≤1400 ^{o}C]/≤100 h
Depressurized	[1200 to 2000 ^{o}C] /100 to 1000h
Thermal Gradient	
Normal conditions	[50 - 200 C^{o}/cm]
Off-normal conditions	[TBD]
Environment	
Normal conditions	He/Ne(in-pile tests)
Off-normal conditions	He; He/H_2O/CO/H_2; He/CO/N_2
Coolant/Impurity Levels	
Normal condition (Design):	Total oxidants [≤7Pa (1ppmv)]
H_2O	[0.07 to 0.7 Pa (0.01 to 0.1 ppmv)]
CO	[1.5 to 6 Pa (0.2 to 8 ppmv)]
CO_2	[0.2 to 1.5 Pa 0.03 to 0.2 ppmv)]
H_2	[3 to 10 Pa (0.5 to 1.5 ppmv)]
CH_4	[0.3 to1.5 Pa (0.5 to 1.5 ppmv)]
N_2	[3 Pa (0.4 ppmv)]

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[*] A lower pressure may be used to satisfy DDN if adequate technical justification is provided.

Pressurized condition	
H_2O	[0 to TBD KPa (0 to TBD atm)]
Depressurized condition	
CO	[0 to 35 KPa (0 to 0.35 atm)]
N_2	[0 to 65 KPa (0 to 0.65 atm)]
Expected failure fraction	
Normal conditions	[10^{-5} to 10^{-3}]
Off-normal conditions	[10^{-4} to 10^{-2}]

2. DESIGNER'S ALTERNATIVES

Alternatives to the acquisition of the above described data are:

2.1 Use the current defective-particle performance models derived from fundamental materials properties and increase design margins to compensate for the large model uncertainties.

2.2 Assume all defective particles fail immediately and design for the predicted higher failure rates; limit the allowable levels of coating defects as necessary to meet fuel failure criteria with the assumption of 100% failure.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Acquire gaseous and metallic fission product release data from TRISO-coated TF particles with missing or defective layers to determine their failure characteristics as a function of environmental conditions, and update the defective particle performance models.

Alternative 2.1 would result in predictive uncertainties in excess of the [10x] predictive accuracy goal. Alternative 2.2 would result in too much conservatism in performance predictions with costly fuel product specification or reactor design changes needed to achieve the required degree of fission product control.

4. SCHEDULE REQUIREMENTS

Preliminary data [6] months before start of Preliminary Design and final data [12] months prior to completion of Final Design.

5. PRIORITY

Urgency: TBD
Cost benefit: M
Uncertainty in existing data: H
Importance of new data: M

6. CONSEQUENCES OF NONEXECUTION AND FALLBACK POSITION

The fallback position is to assume defective TF particles fail immediately. The consequence is expected to be a significantly higher failure fraction prediction than would actually be observed for a given level of as-manufactured defects. Unnecessary conservatism in the performance models will result in unnecessarily stringent fuel product specifications or possible reactor design changes to accommodate unrealistically high, predicted radionuclide levels in the primary circuit.

7. REFERENCES

1. Myers, B.F., "Fuel Design Data Manual," GA Document 901866, Ref. F, General Atomics, August 1987.

2. Ho., F., "Material Models of Pyrocarbon and Pyrolytic Silicon Carbide," GA Document CEGA-002820, Rev. 1, July 1993.

3. Baldwin, C.A., et al., "Fuel Capsule HRB-21 Postirradiation Examination Report," DOE-HTGR-100229, ORNL, April 1995.

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

“AAA” Fuel Program
THERMOCHEMICAL PERFORMANCE DATA FOR DF FUEL
DDN RTW.02.08

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Despite the considerable, high-temperature performance capabilities of TRISO-coated fuel particles, certain thermochemical phenomena have been identified which establish the ultimate thermal performance limits of TRISO particles; these include: (1) kernel migration, (2) fission product/SiC coating interactions, and (3) thermal decomposition of the SiC coating. At least the latter two of these phenomena will occur in TRISO-coated DF particles at sufficiently high temperatures, high thermal gradients and long times; the kinetics of these performance-limiting phenomena need to be quantified for reference DF particles to establish thermochemical performance models for DB-MHR core design and safety analysis.

1.1 Summary of Function/Assumptions

"Retain Radionuclides in Fuel Particles," Assumption: Reference fuel failure models are sufficiently accurate to within a factor of [4] at 95% confidence.

"Protect the Capability to Retain Radionuclides in Fuel Particles," Assumption: Reference fuel failure models are sufficiently accurate to predict failure during core heatup transients to within a factor of [TBD] at 95% confidence.

"Control Transport from Core," Assumption: Adequate data is available to predict fuel performance under transient conditions.

"Retain Radionuclides in Fuel Particles," Assumption: Reference fuel failure models are sufficiently accurate to predict failure under transient conditions to within a factor of 4x at 95% confidence.

1.2 Current Data Base Summary

Kernel migration, the migration of a fuel kernel up a thermal gradient at high temperatures, can eventually cause the failure of the TRISO-coating system if the kernel contacts the SiC coating. This phenomenon is strongly dependent upon the kernel composition and has been observed in U, Th and Pu fuels with both oxide- and carbide-based kernels (e.g., Ref. 1). For kernel migration to occur, both high temperatures and high thermal gradients are required for extended time periods (e.g., kernel migration does not occur even at very high temperatures under core heatup conditions because the thermal gradients are too small).

Kernel migration rates are characterized by a "kernel migration coefficient" (KMC) which is exponentially temperature dependent; these coefficients have been determined for a number of kernels, including UO_2, $(U,Th)C_2$, ThC_2 and ThO_2. To date, significant kernel migration in UCO fuel has not been observed in postirradiation examinations. With plutonium fuels, the kernel migration rate has been shown to be strongly dependent on kernel stoichiometry: particles with a O/Pu ratio = 1.68 showed insignificant kernel migration when irradiated to 70% FIMA at ~1440 ^{o}C in FTE-13 whereas other particles in the same test with a O/Pu ratio = 1.8 migrated excessively (Ref. 2). According to one hypothesis, CO formation is a prerequisite for kernel migration in oxide kernels; consequently, the stoichiometry of UCO and PuO_{2-x} kernels have been tailored to suppress CO formation.

Fission product/SiC interactions appear to occur in all TRISO-coated fuel particles at sufficiently high temperatures and thermal gradients (e.g., Ref. 3). Two classes of fission products have been implicated in the apparent corrosion of the SiC coatings by fission products: (1) lanthanide metals and (2) noble metals, principally Pd. The magnitude of this attack (which probably involves multiple chemical phenomena) appears to be somewhat dependent upon kernel composition. With carbide-based kernels, the lanthanide metals are mobile, and they have been observed to attack the SiC coating on the hotter side of the particle; this type of attack produces a distinct, broad-front reaction zone at the IPyC/SiC interface. With oxide-based kernels, including UCO and PuO_{2-x}, the lanthanides are retained in the kernel as stable oxides, but fission product/SiC reactions are still observed, including in PuO_{2-x} particles; Pd and other noble metals are typically implicated, and the mode of attack appears to be pitting or "worm-holing" in contrast to the broad-front attack by La metals. Large particle-to-particle variability is also observed for each of these corrosion phenomena.

A number of attempts have been made to quantify the rates of fission product/SiC interactions. Originally, a penetration distance was estimated from the observed thinning of SiC coatings during PIE of irradiated fuel specimens. A time-average reaction rate was deduced from this cumulative thinning, and this rate was correlated with a time-average temperature and thermal gradient. In other laboratory tests, a thermal gradient was artificially imposed across irradiated fuel particles, and the cumulative SiC thinning determined as a function of time. Unfortunately, these tests were not performed in such a manner that the functional dependencies of thinning on temperature and thermal gradient could be separated.

Early modeling efforts assumed a uniform particle-to-particle corrosion rate; both linear and square-root time dependencies have been assumed (the Fuel Design Data Manual/Issue E model assumed a linear time dependence with a normal distribution for the SiC thickness). More recent modeling efforts have assumed probability distributions for the reaction rate and SiC thickness with the convolution approximated by a Weibull distribution. This approach is conceptually attractive considering the observed large particle-to-particle variability; however, the resulting correlation (FDDM/F, Ref. 4) gives a time dependence which appears to be nonphysical (a small, but nontrivial, corrosion-induced failure fraction is predicted almost immediately followed by a very modest incremental failure over long time periods).

The thermal decomposition of SiC into its elemental constituents occurs in all TRISO-coated particles at sufficiently high temperatures (>~1800 °C). The rate of thermal decomposition of bare unirradiated SiC has been determined by direct measurements at KFA; however, the applicability of these data to irradiated SiC coatings, especially those protected by intact OPyC layers, is questionable.

More generally, thermal decomposition rates for SiC coatings in TRISO particles have been deduced from postirradiation heating data, specifically from time- and temperature-dependent cesium release data (e.g., Refs. 5 and 6). The underlying assumption is that the fractional cesium release is an indicator of the functional failure of the SiC coatings as a result of thermal decomposition (there is also the embedded assumption that cesium is not diffusively released through intact SiC coatings). Since large particle-to-particle variations are observed in the extent of SiC degradation during postirradiation heating, the assumption is made that the failure rate from thermal decomposition also follows a Weibull distribution. The thermal decomposition rate is projected to be dependent upon the previous irradiation history, but the dominant variable is the postirradiation heating temperature. In many cases, it is difficult to differentiate the effects of burnup and of fast fluence on failure rates from the data.

1.3 Data Needed

Single-effects data are needed to quantify the above phenomena for TRISO-coated Driver Fuel particles under DB-MHR service conditions for normal operation and postulated accidents. These data will be used refine the existing thermochemical performance models for use in core design and safety analysis.

Kernel migration rates are required for TRISO-coated DF particles as a function of temperature, thermal gradient, time and kernel composition; alternatively, it may be demonstrated experimentally that kernel migration is not a performance limiting phenomenon with DF kernels.

The rates of fission product/SiC interactions need to be determined as a function of burnup, temperature, thermal gradient, time and, if appropriate, kernel composition and irradiation history. It is particularly important to separate the temperature and thermal gradient dependencies and to determine the time dependence because these variables are of particular importance in extrapolating the results of relatively short, accelerated irradiation tests to predict in-core performance which is characterized by lower thermal gradients but longer times.

The rates of thermal degradation of irradiated TRISO particles under core heatup conditions need to be determined as a function of burnup, temperature, thermal gradient, time and, if appropriate, kernel composition and irradiation history. The appropriateness of using cesium release as the exclusive indicator of SiC failure needs to be confirmed. The possibility of an enhanced pressure-vessel failure probability as a result of a partially degraded SiC coating also needs to be investigated (as assumed in some German accident fuel performance models).

1.4 Parameters/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications as appropriate:

Parameter	Value
Reference fuel	TRISO-coated DF in compacts
Fuel burnup range	[0 to 85]% FIMA
Peak fast fluence	8 x 10^{25} n/m^2 (E >29 fJ)
System pressure*	
Normal condition	7.1 MPa (70 atm)
Off-normal conditions	0.1 to 7.1 MPa (1 to 70 atm)
Fuel temperatures/time	
Normal condition	[800 to 1400 C]/≤36 months
Off-normal conditions (core conduction cooldown)	
Pressurized (wet)	[≤1400 °C]/≤100 h
Depressurized	[1200 to 2000 °C] /100 to 1000h
Thermal Gradient	
Normal conditions	[50 - 200 C°/cm]
Off-normal conditions	[TBD]
Environment	
Normal conditions	He/Ne(in-pile tests)
Off-normal conditions	He; He/H_2O/CO/H_2; He/CO/N_2
Coolant/Impurity Levels	
Normal condition (Design):	Total oxidants [≤7Pa (1ppmv)]
H_2O	[0.07 to 0.7 Pa (0.01 to 0.1 ppmv)]
CO	[1.5 to 6 Pa (0.2 to 8 ppmv)]
CO_2	[0.2 to 1.5 Pa 0.03 to 0.2 ppmv)]
H_2	[3 to 10 Pa (0.5 to 1.5 ppmv)]
CH_4	[0.3 to1.5 Pa (0.5 to 1.5 ppmv)]
N_2	[3 Pa (0.4 ppmv)]
Pressurized condition	
H_2O	[0 to TBD KPa (0 to TBD atm)]
Depressurized condition	
CO	[0 to 35 KPa (0 to 0.35 atm)]

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[*] A lower pressure may be used to satisfy DDN if adequate technical justification is provided.

N_2	[0 to 65 KPa (0 to 0.65 atm)]
Expected failure fraction	
Normal conditions	[10^{-5} to 10^{-3}]
Off-normal conditions	[10^{-4} to 10^{-2}]

2. DESIGNER'S ALTERNATIVES

Alternatives to the acquisition of the above described data are:

2.1 Use the current thermochemical fuel performance models which have large uncertainties, especially when applied to DF particles.

2.2 Establish more conservative limits on fuel temperatures during normal operation and core heatup accidents based upon existing test data in an attempt to preclude significant temperature-induced particle failure by design.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Alternative 2.1 may result in fuel failure uncertainties for the DB-MHR in excess of the [4x] predictive accuracy goal. Alternative 2.2 would result in excessive conservatism in the core thermal design basis and would likely limit the core thermal power to uneconomically low levels.

4. SCHEDULE REQUIREMENTS

Preliminary data [6] months before start of Preliminary Design and final data [12] months prior to completion of Final Design.

5. PRIORITY

Urgency: TBD
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H

6. CONSEQUENCES OF NONEXECUTION AND FALLBACK POSITION

Alternative 2.1 along with a more conservative fuel and core design to account for the large uncertainties in the current thermochemical performance models when applied to DF particles. Failure to fully exploit the high-temperature performance capability of TRISO-coated DF particles will necessitate a more conservative fuel and core design which could include unnecessarily restrictive limits on fuel temperatures during normal operation and core conduction cooldown transients.

7. REFERENCES

1. Lindemer, T.B., and R.L. Pearson, “Kernel Migration for HTGR Fuels from the Th-U-C-O-N System,” TM-5207, ORNL, April 1976.

2. Miller, C. M., and W. J. Scheffel, "Postirradiation Examination and Evaluation of Peach Bottom FTE-13," GA Document 906939, Rev. N/C, GA Technologies, November 1, 1985.

3. Gruebmeier, H., et al., “Silicon Carbide Corrosion in High-Temperature Gas-Cooled Reactor Fuel Particles,” Nucl Tech., **35**, 413 (1977).

4. Myers, B.F., “Fuel Design Data Manual,” GA Document 901866, Ref. F, General Atomics, August 1987.

5. Nabielek, H., et al., “The Performance of High-Temperature Reactor Fuel Particle at Extreme Temperatures,” Nucl Tech., **84**, 62-81 (1989).

6. Goodin, D.T., “US/FRG Accident Condition Fuel Performance Models,” DOE-HTGR-85107, Rev. 1, General Atomics, 1989.

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

“AAA” Fuel Program
THERMOCHEMICAL PERFORMANCE DATA FOR TF FUEL
DDN RTW.02.09

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Despite the considerable, high-temperature performance capabilities of TRISO-coated fuel particles, certain thermochemical phenomena have been identified which establish the ultimate thermal performance limits of TRISO particles; these include: (1) kernel migration, (2) fission product/SiC coating interactions, and (3) thermal decomposition of the SiC coating. At least the latter two of these phenomena will occur in TRISO-coated TF particles at sufficiently high temperatures, high thermal gradients and long times; the kinetics of these performance-limiting phenomena need to be quantified for reference TF particles to establish thermochemical performance models for DB-MHR core design and safety analysis.

1.1 Summary of Function/Assumptions

"Retain Radionuclides in Fuel Particles," Assumption: Reference fuel failure models are sufficiently accurate to within a factor of [10] at 95% confidence.

"Protect the Capability to Retain Radionuclides in Fuel Particles," Assumption: Reference fuel failure models are sufficiently accurate to predict failure during core heatup transients to within a factor of [TBD] at 95% confidence.

"Control Transport from Core," Assumption: Adequate data is available to predict fuel performance under transient conditions.

"Retain Radionuclides in Fuel Particles," Assumption: Reference fuel failure models are sufficiently accurate to predict failure under transient conditions to within a factor of [10x] at 95% confidence.

1.2 Current Data Base Summary

Kernel migration, the migration of a fuel kernel up a thermal gradient at high temperatures, can eventually cause the failure of the TRISO-coating system if the kernel contacts the SiC coating. This phenomenon is strongly dependent upon the kernel composition and has been observed in U, Th and Pu fuels with both oxide- and carbide-based kernels (e.g., Ref. 1). For kernel migration to occur, both high temperatures and high thermal gradients are required for extended time periods (e.g., kernel migration does not occur even at very high temperatures under core heatup conditions because the thermal gradients are too small).

Kernel migration rates are characterized by a "kernel migration coefficient" (KMC) which is exponentially temperature dependent; these coefficients have been determined for a number of kernels, including UO_2, $(U,Th)C_2$, ThC_2 and ThO_2. To date, significant kernel migration in UCO fuel has not been observed in postirradiation examinations. With plutonium fuels, the kernel migration rate has been shown to be strongly dependent on kernel stoichiometry: particles with a O/Pu ratio = 1.68 showed insignificant kernel migration when irradiated to 70% FIMA at ~1440 ^{o}C in FTE-13 whereas other particles in the same test with a O/Pu ratio = 1.8 migrated excessively (Ref. 2). According to one hypothesis, CO formation is a prerequisite for kernel migration in oxide kernels, and the stoichiometry of UCO and PuO_{2-x} kernels have been tailored to suppress CO formation.

Fission product/SiC interactions appear to occur in all TRISO-coated fuel particles at sufficiently high temperatures and thermal gradients (e.g., Ref. 3). Two classes of fission products have been implicated in the apparent corrosion of the SiC coatings by fission products: (1) lanthanide metals and (2) noble metals, principally Pd. The magnitude of this attack (which probably involves multiple chemical phenomena) appears to be somewhat dependent upon kernel composition. With carbide-based kernels, the lanthanide metals are mobile, and they have been observed to attack the SiC coating on the hotter side of the particle; this type of attack produces a distinct, broad-front reaction zone at the IPyC/SiC interface. With oxide-based kernels, including UCO and PuO_{2-x}, the lanthanides are retained in the kernel as stable oxides, but fission product/SiC reactions are still observed, including in PuO_{2-x} particles; Pd and other noble metals are typically implicated, and the mode of attack appears to be pitting or "worm-holing" in contrast to the broad-front attack by La metals. Large particle-to-particle variability is also observed for each of these corrosion phenomena.

A number of attempts have been made to quantify the rates of fission product/SiC interactions. Originally, a penetration distance was estimated from the observed thinning of SiC coatings during PIE of irradiated fuel specimens. A time-average reaction rate was deduced from this cumulative thinning, and this rate was correlated with a time-average temperature and thermal gradient. In other laboratory tests, a thermal gradient was artificially imposed across irradiated fuel particles, and the cumulative SiC thinning determined as a function of time. Unfortunately, these tests were not performed in such a manner that the functional dependencies of thinning on temperature and thermal gradient could be separated.

Early modeling efforts assumed a uniform particle-to-particle corrosion rate; both linear and square-root time dependencies have been assumed (the Fuel Design Data Manual/Issue E model assumed a linear time dependence with a normal distribution for the SiC thickness). More recent modeling efforts have assumed probability distributions for the reaction rate and SiC thickness with the convolution approximated by a Weibull distribution. This approach is conceptually attractive considering the observed large particle-to-particle variability; however, the resulting correlation (FDDM/F, Ref. 4) gives a time dependence which appears to be nonphysical (a small, but nontrivial, corrosion-induced failure fraction is predicted almost immediately followed by a very modest incremental failure over long time periods).

The thermal decomposition of SiC into its elemental constituents occurs in all TRISO-coated particles at sufficiently high temperatures (>~1800 °C). The rate of thermal decomposition of bare unirradiated SiC has been determined by direct measurements at KFA; however, the applicability of these data to irradiated SiC coatings, especially those protected by intact OPyC layers, is questionable.

More generally, thermal decomposition rates for SiC coatings in TRISO particles have been deduced from postirradiation heating data, specifically from time- and temperature-dependent cesium release data (e.g., Refs. 5 and 6). The underlying assumption is that the fractional cesium release is an indicator of the functional failure of the SiC coatings as a result of thermal decomposition (there is also the embedded assumption that cesium is not diffusively released through intact SiC coatings). Since large particle-to-particle variations are observed in the extent of SiC degradation during postirradiation heating, the assumption is made that the failure rate from thermal decomposition also follows a Weibull distribution. The thermal decomposition rate is projected to be dependent upon the previous irradiation history, but the dominant variable is postirradiation heating temperature. In many cases, it is difficult to differentiate the effects of burnup and of fast fluence on failure from the data.

1.3 Data Needed

Single-effects data are needed to quantify the above phenomena for TRISO-coated Transmutation Fuel particles under DB-MHR service conditions for normal operation and postulated accidents. These data will be used refine the existing thermochemical performance models for use in core design and safety analysis.

Kernel migration rates are required for TRISO-coated TF particles as a function of temperature, thermal gradient, time and kernel composition; alternatively, it may be demonstrated experimentally that kernel migration is not a performance limiting phenomenon with TF kernels.

The rates of fission product/SiC interactions need to be determined as a function of burnup, temperature, thermal gradient, time and, if appropriate, kernel composition and irradiation history. It is particularly important to separate the temperature and thermal gradient dependencies and to determine the time dependence because these variables are of particular importance in extrapolating the results of relatively short, accelerated irradiation tests to predict in-core performance which is characterized by lower thermal gradients but longer times.

The rates of thermal degradation of irradiated TRISO particles under core heatup conditions need to be determined as a function of burnup, temperature, thermal gradient, time and, if appropriate, kernel composition and irradiation history. The appropriateness of using cesium release as the exclusive indicator of SiC failure needs to be confirmed. The possibility of an enhanced pressure-vessel failure probability as a result of a partially degraded SiC coating also needs to be investigated (as assumed in some German accident fuel performance models).

Transmutation Fuel with its high concentrations of minor actinides can be considered an exotic TRISO fuel by traditional, coated-particle fuel standards; consequently, the high temperature performance data generated for TF will need to be carefully evaluated to confirm that no new thermochemical failure mechanisms occur in addition to the above mechanisms. This requirement will mandate comprehensive postirradiation examinations, especially for the margin tests which will proceed until some particle failure occurs.

1.4 Parameters/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications as appropriate:

Parameter	Value
Reference fuel	TRISO-coated TF in compacts
Fuel burnup range	[0 to 85]% FIMA
Peak fast fluence	8 x 10^{25} n/m^2 (E >29 fJ)
System pressure[*]	
Normal condition	7.1 MPa (70 atm)
Off-normal conditions	0.1 to 7.1 MPa (1 to 70 atm)
Fuel temperatures/time	
Normal condition	[800 to 1400 C]/≤36 months
Off-normal conditions (core conduction cooldown)	
Pressurized (wet)	[≤1400 °C]/≤100 h
Depressurized	[1200 to 2000 °C] /100 to 1000h
Thermal Gradient	
Normal conditions	[50 - 200 C°/cm]
Off-normal conditions	[TBD]
Environment	
Normal conditions	He/Ne(in-pile tests)
Off-normal conditions	He; He/H_2O/CO/H_2; He/CO/N_2
Coolant/Impurity Levels	
Normal condition (Design):	Total oxidants [≤7Pa (1ppmv)]
H_2O	[0.07 to 0.7 Pa (0.01 to 0.1 ppmv)]
CO	[1.5 to 6 Pa (0.2 to 8 ppmv)]
CO_2	[0.2 to 1.5 Pa 0.03 to 0.2 ppmv)]
H_2	[3 to 10 Pa (0.5 to 1.5 ppmv)]
CH_4	[0.3 to1.5 Pa (0.5 to 1.5 ppmv)]

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[*] A lower pressure may be used to satisfy DDN if adequate technical justification is provided.

N_2	[3 Pa (0.4 ppmv)]
Pressurized condition	
H_2O	[0 to TBD KPa (0 to TBD atm)]
Depressurized condition	
CO	[0 to 35 KPa (0 to 0.35 atm)]
N_2	[0 to 65 KPa (0 to 0.65 atm)]
Expected failure fraction	
Normal conditions	[10^{-5} to 10^{-3}]
Off-normal conditions	[10^{-4} to 10^{-2}]

2. DESIGNER'S ALTERNATIVES

Alternatives to the acquisition of the above described data are:

2.1 Use the current thermochemical fuel performance models which have large uncertainties, especially when applied to TF particles.

2.2 Establish more conservative limits on fuel temperatures during normal operation and core heatup accidents based upon existing test data in an attempt to preclude significant temperature-induced particle failure.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Alternative 2.1 may result in fuel failure uncertainties for the DB-MHR in excess of the [10x] predictive accuracy goal. Alternative 2.2 would result in excessive conservatism in the core thermal design basis and would likely limit the core thermal power to uneconomically low levels.

4. SCHEDULE REQUIREMENTS

Preliminary data [6] months before start of Preliminary Design and final data [12] months prior to completion of Final Design.

5. PRIORITY

Urgency: TBD
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H

6. CONSEQUENCES OF NONEXECUTION AND FALLBACK POSITION

Alternative 2.1 along with a more conservative fuel and core design to account for the large uncertainties in the current thermochemical performance models. Failure to fully exploit the high-temperature performance capability of TRISO-coated TF particles will necessitate a more conservative fuel and core design which could include unnecessarily restrictive limits on fuel temperatures during normal operation and core conduction cooldown transients.

7. REFERENCES

1. Lindemer, T.B., and R.L. Pearson, "Kernel Migration for HTGR Fuels from the Th-U-C-O-N System," TM-5207, ORNL, April 1976.

2. Miller, C. M., and W. J. Scheffel, "Postirradiation Examination and Evaluation of Peach Bottom FTE-13," GA Document 906939, Rev. N/C, GA Technologies, November 1, 1985.

3. Gruebmeier, H., et al., "Silicon Carbide Corrosion in High-Temperature Gas-Cooled Reactor Fuel Particles," Nucl Tech. **35**, 413 (1977).

4. Myers, B.F., "Fuel Design Data Manual," GA Document 901866, Ref. F, General Atomics, August 1987.

5. Nabielek, H., et al., "The Performance of High-Temperature Reactor Fuel Particle at Extreme Temperatures," Nucl Tech. **84**, 62-81 (1989).

6. Goodin, D.T., "US/FRG Accident Condition Fuel Performance Models," DOE-HTGR-85107, Rev. 1, General Atomics, 1989.

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

"AAA" Fuel Program
FUEL COMPACT THERMOPHYSICAL PROPERTIES
DDN RTW.02.10

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The thermophysical properties (e.g., thermal conductivity, heat capacity, irradiation-induced dimensional change, etc.) of unirradiated and irradiated reference Driver Fuel and Transmutation Fuel compacts need to be characterized to support the validation of the core thermal design methods.

1.1 Summary of Functions/Assumptions

[TBD]

1.2 Current Data Base Summary

Data have been obtained from measurements of thermal expansion and conductivity and heat capacity for HTGR fuel compacts at General Atomics from previous commercial MHR programs (Ref. 1). Other tests have been performed on irradiated fuel compacts to determine thermal conductivity versus macroporosity. However, this database is small and scattered. Likewise, correlations for the irradiation-induced dimensional change of petroleum pitch based fuel compacts have developed as a function of shim content, fast fluence, and temperature (Ref. 1).

1.3 Data Needed

Thermophysical and mechanical properties data are needed for a statistically significant number of resin-based DF and TF compacts including heat capacity, thermal expansion, thermal conductivity, and irradiation-induced dimensional change. Data are required as a function of shim content, fast neutron fluence and temperature to validate the core thermal design methods at a 95% statistical confidence level.

The data need to be obtained from both unirradiated and irradiated fuel compacts containing TRISO-coated DF and TF particles. Simulated fuel compacts containing inert TRISO-coated particles rather than fuel particles may be used. An in-pile thermal conductivity experiment may be required. In addition, the effect of off-normal conditions on thermal conductivity of compacts also needs to be determined.

1.4 Parameters/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications as appropriate:

Parameter	Value
Test article	TRISO-coated DF and TF in compacts
Compact attributes	
Dimensions	12.5 mm dia. x 50 mm long
Matrix type	[thermosetting resin]
Packing fraction range, %	[TBD]
Peak fast fluence	8×10^{25} n/m^2 (E >29 fJ)
System pressure[*]	
Normal condition	7.1 MPa (70 atm)
Off-normal conditions	0.1 to 7.1 MPa (1 to 70 atm)
Fuel temperatures/time	
Normal condition	[800 to 1400 C]/≤36 months
Off-normal conditions (core conduction cooldown)	
Pressurized (wet)	[≤1400 °C]/≤100 h
Depressurized	[1200 to 2000 °C] /100 to 1000h
Thermal Gradient	
Normal conditions	[50 - 200 C°/cm]
Off-normal conditions	[TBD]
Environment	
Normal conditions	He/Ne(in-pile tests)
Off-normal conditions	He; He/H_2O/CO/H_2; He/CO/N_2
Coolant/Impurity Levels	
Normal condition (Design):	Total oxidants [≤7Pa (1ppmv)]
H_2O	[0.07 to 0.7 Pa (0.01 to 0.1 ppmv)]
CO	[1.5 to 6 Pa (0.2 to 8 ppmv)]
CO_2	[0.2 to 1.5 Pa 0.03 to 0.2 ppmv)]
H_2	[3 to 10 Pa (0.5 to 1.5 ppmv)]
CH_4	[0.3 to1.5 Pa (0.5 to 1.5 ppmv)]
N_2	[3 Pa (0.4 ppmv)]
Pressurized condition	
H_2O	[0 to TBD KPa (0 to TBD atm)]
Depressurized condition	
CO	[0 to 35 KPa (0 to 0.35 atm)]
N_2	[0 to 65 KPa (0 to 0.65 atm)]

2. DESIGNER'S ALTERNATIVES

Alternatives to the acquisition of the above data are:

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[*] A lower pressure may be used to satisfy DDN if adequate technical justification is provided.

2.1 Use the existing database, and add design margins to account for the uncertainties.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected approach is to generate thermophysical properties data, including thermal expansion, conductivity and heat capacity, for a statistically significant number of DF and TF compacts made of prototypical materials by the reference compacting process.

4. SCHEDULE REQUIREMENTS

Final data needed 12 months prior to completion of Final Design.

5. PRIORITY

Urgency: TBD
Cost benefit: L
Uncertainty in existing data: M
Importance of new data: L

6. FALLBACK POSITION AND CONSEQUENCES ON NONEXECUTION

The fallback position is to rely on Alternative 2.1. Existing thermal properties data from completed programs are available and will have to be extrapolated to determine the effects on fuel materials under different irradiation conditions. Consequently, this will result in extra uncertainty and/or conservatism in the thermal design and in the calculated fuel temperatures.

7. REFERENCES

1. Myers, B.F., “Fuel Design Data Manual,” GA Document 901866, Ref. F, General Atomics, August 1987.

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

“AAA” Fuel Program
NORMAL-OPERATION DF PERFORMANCE VALIDATION DATA
DDN RTW.02.11

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The failure of reference Driver Fuel particles in compacts, must be limited to low fractional values in order to meet the core release criteria required for the DB-MHR. The integrated methods and codes (e.g., SURVEY) used to predict failure of reference DF particles under normal irradiation conditions must be validated to have the specified predictive accuracies. Integral test data, representative of the DB-MHR, are required for comparison with code predictions. These validation data must be independent of that used to develop the predictive methods.

1.1 Summary of Function/Assumptions

"Retain Radionuclides in Fuel Particles," Assumption: Reference fuel failure models are accurate to within a factor of [4] at 95% confidence.

"Control Transport from Core," Assumption: Validated methods and data are available to adequately assess fuel failure, fission product transport, and release from the primary coolant circuit.

1.2 Current Data Base Summary

The reference performance models for high-burnup PuO_{2-x} fuel are based largely on previous experience with TRISO-coated, HEU carbide fuels, ThO_2, UO_2 and, to a lesser extent, LEU UCO fuel (Ref. 1). There are limited irradiation performance data (e.g., kernel migration distances) for TRISO-coated Pu fuels with several kernel compositions which were considered when developing these Pu models (Ref. 2).

The irradiation performance of $(Th,U)C_2$ and ThC_2 TRISO fuel particles in Fort St. Vrain was apparently well predicted since the noble gas release rates were predicted to within a factor of 2x (Ref. 3); however, no destructive PIE of FSV fuel elements was performed at EOL so the evidence is circumstantial. For high-defect, LEU UCO/ThO_2 particles with 4- and 5-layer TRISO coatings, the available capsule irradiation data show a range of about 10x between observed and predicted gas release at peak exposure. However, the failure rates of 7-layer and 8-layer TRISO-P fuel in HRB-21 and in the three NPR capsules, respectively, were grossly under predicted (Ref. 4).

No performance models have been developed to date for plutonium fuels containing significant quantities of Np.

1.3 Data Needed

Integral test data are needed to characterize the irradiation performance of reference Driver Fuel particles in fuel compacts under bounding DB-MHR core service conditions. These data are needed to provide a experimental basis for validation of fuel performance models to be used in the SURVEY code to predict DF behavior under irradiation conditions.

The complete characterization of the fuel samples prior to irradiation is needed. The fission gas release and temperature as a function of time during each irradiation test is required to determine in-pile performance. After irradiation, the samples need to be well-characterized. Dimensions, physical integrity, and microstructure, phases and chemical composition of phases in kernels and fission product distributions as determined by metallography and electron microscopy will be determined as needed. The thermal and nuclear histories of each sample are needed. These data will be used to determine failure fraction, fission product retention and failure mechanisms of particles and the irradiation behavior of TRISO-coated DF particles.

These data are needed to validate that the observed failure at 95% confidence is within [4x] of the 50% confidence failure predicted by fuel performance methods which are used in core design. [Quality assurance for the test programs must be in accordance with the requirements for validation testing as defined by NQA-1-1994.]

1.4 Parameters/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications as appropriate:

The fuel shall be irradiated primarily to normal peak conditions, but some testing shall done at core-average service conditions. In addition, the fuel will be irradiated to beyond-normal conditions to quantify performances margins (i.e., fuel will be irradiated at high temperature and/or to high fast fluence until on-line R/B measurements indicate a failure fraction in the range of [10^{-3} to 10^{-2}]). The test conditions are the following:

Parameter	Value
Reference fuel	TRISO-coated DF in compacts
Fuel burnup range	[0 to 85]% FIMA
Peak fast fluence	8×10^{25} n/m^2 (E >29 fJ)
System pressure*	7.1 MPa (70 atm)
Fuel temperatures/time	[800 to 1400 C]/≤ 36 months

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*A lower pressure may be used to satisfy DDN if adequate technical justification is provided.

Thermal Gradient	[50 - 200 C^o/cm]
Environment	He/Ne(in-pile tests)
Coolant/Impurity Levels	
Normal operation (Design):	Total oxidants [≤7Pa (1ppmv)]
H_2O	[0.07 to 0.7 Pa (0.01 to 0.1 ppmv)]
CO	[1.5 to 6 Pa (0.2 to 8 ppmv)]
CO_2	[0.2 to 1.5 Pa 0.03 to 0.2 ppmv)]
H_2	[3 to 10 Pa (0.5 to 1.5 ppmv)]
CH_4	[0.3 to1.5 Pa (0.5 to 1.5 ppmv)]
N_2	[3 Pa (0.4 ppmv)]
Expected failure fraction	[10^{-5} to 10^{-3}]

2. DESIGNER'S ALTERNATIVES

Alternatives to the acquisition of the above described data are:

2.1 Argue that the reference PuO_{2-x} performance models, which are based largely on U and Th data and a small amount of Pu data, are applicable to Driver Fuel.

2.2 Extrapolate FRG fuel performance data - largely for 10%-enriched UO_2 in pebbles - and models, and argue they can be applied to Driver Fuel in compacts without further validation.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Initially, on-line fission product release data and PIE data will be obtained for TRISO-coated DF fuel particles in fuel compacts made on developmental fuel fabrication equipment and irradiated under accelerated conditions ([TBD] degree of acceleration). Subsequently, these data will be confirmed by limited testing of reference fuel made on prototypical, full-scale manufacturing equipment and irradiated under both accelerated and near real-time conditions. These data will be used to validate the normal condition fuel performance methods for Driver Fuel.

Alternatives 2.1 and 2.2 were not chosen because of the high probability that formal validation will be required for reference Driver Fuel in compacts, and that extended delays in safety review would result from failure to get explicit data on reference fuel (i.e., Alternatives 2.1 and 2.2 are not considered technically credible). In addition the FRG fuel performance data would suffer additional risk of not having a formal NQA-1 pedigree.

4. SCHEDULE REQUIREMENTS

The data for validation of normal condition performance models must be obtained 12 months prior to completion of Final Design.

5. PRIORITY

Urgency: TBD
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H

6. CONSEQUENCES OF NONEXECUTION AND FALLBACK POSITION

The fallback position would be to argue that the no explicit validation data on Driver Fuel performance are needed due to the existing international data base for TRISO-coated fuels. The designer would rely on existing TRISO performance data for validating the normal condition fuel performance methods and argue kernel composition and burnups are second-order effects. The consequences are expected to consist of (1) significant difficulties and delays in safety review while technical arguments of the similarities of various TRISO-coated fuels are debated and (2) increased risk that the performance of the initial core will not be as good as predicted because the performance methods under predict failure.

This fallback position is considered marginally viable only in combination with significantly relaxed limits on fuel failure during normal operation and the inclusion of a high-pressure containment and active cooling systems in the design and the imposition of stringent technical specifications for plant operation, including a comprehensive reactor surveillance program.

7. REFERENCES

1. “MHTGR Plutonium Consumption Study, Phase II Extension FY-94 Final Report,” GA/DOE-156-94, General Atomics, September 30, 1994.

2. “MHTGR Plutonium Consumption Study, Phase II Final Report,” GA/DOE-051-94, General Atomics, April 29, 1994.

3. Baxter, A.M., et al., “FSV Experience in Support of the GT-MHR Reactor Physics, Fuel Performance, and Graphite,” GA-A21925, General Atomics, November 1994.

4. Leikind, B.J., et al., “MHTGR TRISO-P Fuel Failure Evaluation Report,” DOE-HTGR-90390, General Atomics, October 1993.

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

“AAA” Fuel Program
NORMAL-OPERATION TF PERFORMANCE VALIDATION DATA
DDN RTW.02.12

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The failure of reference Transmutation Fuel particles in compacts, must be limited to low fractional values in order to meet the core release criteria required for the DB-MHR. The integrated methods and codes (e.g., SURVEY) used to predict failure of reference TF particles under normal irradiation conditions must be validated to have the specified predictive accuracies. Integral test data, representative of the DB-MHR, are required for comparison with code predictions. These validation data must be independent of that used to develop the predictive methods.

1.1 Summary of Function/Assumptions

"Retain Radionuclides in Fuel Particles," Assumption: Reference fuel failure models are accurate to within a factor of [10] at 95% confidence.

"Control Transport from Core," Assumption: Validated methods and data are available to adequately assess fuel failure, fission product transport, and release from the primary coolant circuit.

1.2 Current Data Base Summary

The reference performance models for high-burnup PuO_{2-x} fuel are based largely on previous experience with TRISO-coated, HEU carbide fuels, ThO_2, UO_2 and, to a lesser extent, LEU UCO fuel (Ref. 1). There are limited irradiation performance data (e.g., kernel migration distances) for TRISO-coated Pu fuels with several kernel compositions which were considered when developing these Pu models (Ref. 2).

The irradiation performance of $(Th,U)C_2$ and ThC_2 TRISO fuel particles in Fort St. Vrain was apparently well predicted since the noble gas release rates were predicted to within a factor of 2x (Ref. 3); however, no destructive PIE of FSV fuel elements was performed at EOL so the evidence is circumstantial. For high-defect, LEU UCO/ThO_2 particles with 4- and 5-layer TRISO coatings, the available capsule irradiation data show a range of about 10x between observed and predicted gas release at peak exposure. However, the failure rates of 7-layer and 8-layer TRISO-P fuel in HRB-21 and in the three NPR capsules, respectively, were grossly under predicted (Ref. 4).

No performance models have been developed to date for plutonium fuels containing significant quantities of Minor Actinides.

1.3 Data Needed

Integral test data are needed to characterize the irradiation performance of reference Transmutation Fuel particles in compacts under bounding DB-MHR core service conditions. These data are needed to provide a experimental basis for validation of fuel performance models to be used in the SURVEY code to predict TF behavior under irradiation conditions.

The complete characterization of the fuel samples prior to irradiation is needed. The fission gas release and temperature as a function of time during each irradiation test is required to determine in-pile performance. After irradiation, the samples need to be well-characterized. Dimensions, physical integrity, and microstructure, phases and chemical composition of phases in kernels and fission product distributions as determined by metallography and electron microscopy will be determined as needed. The thermal and nuclear histories of each sample are needed. These data will be used to determine failure fraction, fission product retention and failure mechanisms of particles and the irradiation behavior of TRISO-coated TF particles.

These data are needed to validate that the observed failure at 95% confidence is within [10x] of the 50% confidence failure predicted by fuel performance methods which are used in core design.

1.4 Parameters/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications as appropriate:

The fuel shall be irradiated primarily to normal peak conditions, but some testing shall done at core-average service conditions. In addition, the fuel will be irradiated to beyond-normal conditions to quantify performances margins (i.e., fuel will be irradiated at high temperature and/or to high fast fluence until on-line R/B measurements indicate a failure fraction in the range of [10^{-3} to 10^{-2}]). The test conditions are the following:

Parameter	Value
Reference fuel	TRISO-coated TF in compacts
Fuel burnup range	[0 to 85]% FIMA
Peak fast fluence	8×10^{25} n/m^2 (E >29 fJ)
System pressure[*]	7.1 MPa (70 atm)
Fuel temperatures/time	[800 to 1400 C]/≤36 months
Thermal Gradient	[50 - 200 C^o/cm]
Environment	He/Ne(in-pile tests)
Coolant/Impurity Levels	
Normal operation (Design):	Total oxidants [≤7Pa (1ppmv)]
H_2O	[0.07 to 0.7 Pa (0.01 to 0.1 ppmv)]

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[*] A lower pressure may be used to satisfy DDN if adequate technical justification is provided.

CO	[1.5 to 6 Pa (0.2 to 8 ppmv)]
CO_2	[0.2 to 1.5 Pa 0.03 to 0.2 ppmv)]
H_2	[3 to 10 Pa (0.5 to 1.5 ppmv)]
CH_4	[0.3 to1.5 Pa (0.5 to 1.5 ppmv)]
N_2	[3 Pa (0.4 ppmv)]
Expected failure fraction	[10^{-5} to 10^{-3}]

2. DESIGNER'S ALTERNATIVES

Alternatives to the acquisition of the above described data are:

2.1 Argue that the reference PuO_{2-x} performance models, which are based largely on U and Th data and a small amount of Pu data, are applicable to Transmutation Fuel.

2.2 Extrapolate FRG fuel performance data - largely for 10%-enriched UO_2 in pebbles - and models, and argue they can be applied to Transmutation Fuel in compacts without further validation.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Initially, on-line fission product release data and PIE data will be obtained for TRISO-coated TF fuel particles in fuel compacts made on developmental fuel fabrication equipment and irradiated under accelerated conditions ([TBD] degree of acceleration). Subsequently, these data will be confirmed by limited testing of reference fuel made on prototypical, full-scale manufacturing equipment and irradiated under both accelerated and near real-time conditions. These data will be used to validate the normal condition fuel performance methods for Transmutation Fuel.

Alternatives 2.1 and 2.2 were not chosen because of the high probability that formal validation will be required for reference Transmutation Fuel in compacts, and that extended delays in safety review would result from failure to get explicit data on reference fuel (i.e., Alternatives 2.1 and 2.2 are not considered technically credible). In addition the FRG fuel performance data would suffer additional risk of not having a formal NQA-1 pedigree.

4. SCHEDULE REQUIREMENTS

The data for validation of normal condition performance models must be obtained 12 months prior to completion of Final Design.

5. PRIORITY

Urgency: TBD
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H

6. CONSEQUENCES OF NONEXECUTION AND FALLBACK POSITION

The fallback position would be to argue that the no explicit validation data on Transmutation Fuel performance is needed due to the existing international data base for TRISO-coated fuels. The designer would rely on existing TRISO performance data for validating the normal condition fuel performance methods and argue kernel composition and burnups are second-order effects. The consequences are expected to consist of (1) significant difficulties and delays in safety review while technical arguments of the similarities of various TRISO-coated fuels are debated and (2) increased risk that the performance of the initial core will not be as good as predicted because the performance methods under predict failure.

This fallback position might be considered marginally viable in combination with significantly relaxed limits on fuel failure during normal operation and the inclusion of a high-pressure containment and active cooling systems in the design and the imposition of stringent technical specifications for plant operation, including a comprehensive reactor surveillance program.

7. REFERENCES

1. “MHTGR Plutonium Consumption Study, Phase II Extension FY-94 Final Report,” GA/DOE-156-94, General Atomics, September 30, 1994.

2. “MHTGR Plutonium Consumption Study, Phase II Final Report,” GA/DOE-051-94, General Atomics, April 29, 1994.

3. Baxter, A.M., et al., “FSV Experience in Support of the GT-MHR Reactor Physics, Fuel Performance, and Graphite,” GA-A21925, General Atomics, November 1994.

4. Leikind, B.J., et al., “MHTGR TRISO-P Fuel Failure Evaluation Report,” DOE-HTGR-90390, General Atomics, October 1993.

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

“AAA” Fuel Program
ACCIDENT DF PERFORMANCE VALIDATION DATA
DDN RTW.02.13

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The failure of reference Driver Fuel particles in compacts, must be limited to low fractional values in order to meet the core release criteria required for the DB-MHR [with a Vented, Low-Pressure Containment (VLPC)]. The integrated methods and codes (e.g., SORS) used to predict failure of reference DF particles under postulated accident conditions must be validated to have the specified predictive accuracies. Integral test data, representative of postulated DB-MHR accidents, are required for comparison with code predictions. These validation data must be independent of the data originally used to develop the predictive methods.

1.1 Summary of Function/Assumptions

"Protect the Capability to Retain Radionuclides in Fuel Particles," Assumption: Reference fuel failure models are sufficiently accurate to predict failure during core heatup transients to within a factor of [TBD] at 95% confidence.

"Control Transport from Core," Assumption: Adequate data is available to predict fuel performance under transient conditions.

"Retain Radionuclides in Fuel Particles," Assumption: Reference fuel failure models are sufficiently accurate to predict failure under transient conditions to within factor of [4] at 95% confidence.

1.2 Current Data Base Summary

There are no postirradiation heating data for TRISO-coated PuO_x fuel particles. TRISO coated, 500-µm PuO_x kernels, which were diluted with carbon (C/PuO_2 = ~20), were irradiated in the Studsvik R2 materials test reactor and the Dragon HTR at high temperature (1450 and 1275 ^{o}C, respectively), and low on-line fission gas release was measured (Ref. 1). Two of the four tests in Studsvik were thermally cycled to 1850 ^{o}C to simulate accident conditions, and an increase in fission gas release was observed [no quantitative results available at this writing]. TRISO coated, PuO_{2-x} and $(Th,Pu)O_{2-x}$ particles were irradiated in Peach Bottom fuel test element FTE-13 at temperatures up to 1440 ^{o}C, but no postirradiation heating tests were performed (Ref. 2).

Prior US data taken in support of fuel performance models to predict accident behavior were primarily obtained from tests with unbonded, TRISO-coated U and Th particles rather than with compacts, characterized by relatively small sample sizes, and with a variety of fuel types heated in dry helium (e.g., Ref. 3). There is a paucity of US data in the 1400 to 1800 °C temperature range because most measurements were made at higher temperatures characteristic of large HTGR core heatup accidents; moreover, these test specimens typically had relatively high levels of as-manufactured heavy-metal contamination and coating defects.

There is a robust German data base for the performance of TRISO-coated, HEU $(Th,U)O_2$ and LEU UO_2 particles in fuel spheres under dry heating conditions; these fuel particles have performed superbly at temperatures up to 1600 °C (e.g., Ref. 4). There are also some German data for heating at higher temperatures and in the presence of high moisture which indicate degraded performance.

First-order methods have been developed, including a joint U.S./FRG accident condition model based on both U.S. and FRG dry heating data (Ref. 3). A complete uncertainty analysis for this methodology has not been conducted, but the uncertainty in the model has been estimated to be ~12x at the 95% confidence level for the FRG fuel in the range of conditions of interest. The wet heating data have not been modeled.

1.3 Data Needed

Data are needed to determine failure of TRISO-coated DF particles in fuel compacts as indicated by the release of fission products, such as Cs-137, from irradiated fuel compacts under transient conditions characteristic of pressurized and depressurized conduction cooldown events in the DB-MHR. While it is known that fuel compact matrix and OPyC coatings will oxidize under air and moisture ingress conditions, validation of the methods used to predict the rate of corrosion is needed. Furthermore, the corrosion rate and subsequent failure of the SiC under high temperature high moisture conditions is not known for irradiated fuel particles, so data are needed to make predictions on TRISO particle failure under high moisture conditions.

The data must validate that the physical models are (a) suitable for use in design codes (e.g., SORS) covering the range of expected service conditions, (b) capable of predicting the failure of high-burnup DF fuel particles to within [4x] at 95% confidence (fuel failure to be inferred from measured fission product release and PIE). [Quality assurance for the test programs must be in accordance with the requirements for validation testing as defined by NQA-1-1994.]

1.4 Parameters/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications:

The irradiated DF fuel shall be heated primarily peak core heatup conditions (i.e., 1600 ^{o}C), but some testing shall done at core-average conditions. In addition, the fuel will be heated to beyond reactor conditions to quantify performances margins (i.e., fuel will be heated at high temperature (>1600 ^{o}C) until on-line fission gas release measurements indicate a failure fraction in the range of [10^{-3} to 10^{-2}]). The test conditions are the following:

:

Fuel First Exposed to Normal Operating Conditions

Parameter	Value
Reference fuel	TRISO-coated DF in compacts
Fuel burnup range	[0 to 85]% FIMA
Peak fast fluence	8 x 10^{25} n/m^{2} (E >29 fJ)
System pressure[*]	7.1 MPa (70 atm)
Fuel temperatures/time	[800 to 1400 C]/≤36 months
Thermal Gradient	[50 - 200 C^{o}/cm]
Environment	He/Ne(in-pile tests)
Coolant/Impurity Levels	
Normal operation (Design):	Total oxidants [≤7Pa (1ppmv)]
H_2O	[0.07 to 0.7 Pa (0.01 to 0.1 ppmv)]
CO	[1.5 to 6 Pa (0.2 to 8 ppmv)]
CO_2	[0.2 to 1.5 Pa 0.03 to 0.2 ppmv)]
H_2	[3 to 10 Pa (0.5 to 1.5 ppmv)]
CH_4	[0.3 to1.5 Pa (0.5 to 1.5 ppmv)]
N_2	[3 Pa (0.4 ppmv)]
Expected failure fraction	[10^{-5} to 10^{-3}]

Off-Normal Conditions

Reference transients	Pressurized conduction cooldown with and without water ingress Depressurized conduction
Environment	He; He/CO/H_2; He/CO/N_2
Fuel temperatures/time	
Pressurized	[1300 ^{o}C]/up to 100 h
Depressurized	[1200 to 1600 ^{o}C] from 300 to 1000 h
	[1600 to 2000 ^{o}C] from 100 to 200 h
System pressure	0.1 to 7.1 MPa (1 to 70 atm)
Coolant Impurity Levels	
Pressurized conditions	[0 to TBD MPa (0 to TBD atm) H_2O]
Depressurized	[0 to 35 KPa (0 to 0.35 atm)] CO
	[0 to 65 KPa (0 to 0.65 atm)] N_2

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[*] A lower pressure may be used to satisfy DDN if adequate technical justification is provided.

2. DESIGNER'S ALTERNATIVES

Alternatives to the acquisition of the above described data are:

2.1 Utilize existing accident-condition fuel performance data, primarily from heating tests with German LEU UO_2 particles in pebbles in the 1200 to 2000 ^{o}C temperature range; abandon the VLPC, and add a high-pressure containment and active cooling systems to the design.

2.2 Utilize FRG accident-condition models and argue that these data are directly applicable to reference U.S. fuel; abandon the VLPC, and add a high-pressure containment and active cooling systems the design.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Fission product release data from irradiated DF compacts will be obtained as a function of time and temperatures simulating core conduction cooldown transients. These data will allow the design methods describing performance of fuel under conduction cooldown conditions to be validated and the uncertainty interval reduced to the required accuracy.

Alternatives 2.1 and 2.2 were not chosen because of the high probability that formal validation with explicit accident condition data for the reference DF will be required, and that significant difficulties in safety review would result (i.e., Alternatives 2.1 and 2.2 are not considered technically viable for the licensing of a VLPC). In addition the FRG fuel performance data would suffer additional risk of not complying with NQA-1 requirements.

4. SCHEDULE REQUIREMENTS

Validated design methods reflecting the expanded data base are needed 12 months prior to completion of Final Design.

5. PRIORITY

Urgency: TBD
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H

6. CONSEQUENCES OF NONEXECUTION AND FALLBACK POSITION

The fallback position is to rely upon existing fuel performance data and models and to replace the VLPC with a high-pressure containment and active cooling systems. Even with a high-pressure containment, stringent technical specifications on core operation and on containment performance should be anticipated. While such a design might be licensable, the investment risk would be considerable, and the economic viability of this design approach is questionable.

7. REFERENCES

1. "MHTGR Plutonium Consumption Study, Phase II Final Report," GA/DOE-051-94, General Atomics, April 29, 1994.

2. Miller, C. M., and W. J. Scheffel, "Postirradiation Examination and Evaluation of Peach Bottom FTE-13," GA Document 906939, Rev. N/C, GA Technologies, November 1, 1985.

3. Goodin, D.T., "US/FRG Accident Condition Fuel Performance Models," DOE-HTGR-85107, Rev. 1, General Atomics, 1989.

4. Nabielek, H., et al., "The Performance of High-Temperature Reactor Fuel Particle at Extreme Temperatures," Nucl Tech. **84**, 62-81 (1989).

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

"AAA" Fuel Program
ACCIDENT TF PERFORMANCE VALIDATION DATA
DDN RTW.02.14

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The failure of reference Transmutation Fuel particles in compacts, must be limited to low fractional values in order to meet the core release criteria required for the DB-MHR [with a Vented, Low-Pressure Containment (VLPC)]. The integrated methods and codes (e.g., SORS) used to predict failure of reference DF particles under postulated accident conditions must be validated to have the specified predictive accuracies. Integral test data, representative of postulated DB-MHR accidents, are required for comparison with code predictions. These validation data must be independent of the data originally used to develop the predictive methods.

1.1 Summary of Function/Assumptions

"Protect the Capability to Retain Radionuclides in Fuel Particles," Assumption: Reference fuel failure models are sufficiently accurate to predict failure during core heatup transients to within a factor of [TBD] at 95% confidence.

"Control Transport from Core," Assumption: Adequate data is available to predict fuel performance under transient conditions.

"Retain Radionuclides in Fuel Particles," Assumption: Reference fuel failure models are sufficiently accurate to predict failure under transient conditions to within factor of [10] at 95% confidence.

1.2 Current Data Base Summary

There are no postirradiation heating data for TRISO-coated PuO_x fuel particles. TRISO coated, 500-µm PuO_x kernels, which were diluted with carbon (C/PuO_2 = ~20), were irradiated in the Studsvik R2 materials test reactor and the Dragon HTR at high temperature (1450 and 1275 °C, respectively), and low on-line fission gas release was measured (Ref. 1). Two of the four tests in Studsvik were thermally cycled to 1850 °C to simulate accident conditions, and an increase in fission gas release was observed [no quantitative results available at this writing]. TRISO coated, PuO_{2-x} and $(Th,Pu)O_{2-x}$ particles were irradiated in Peach Bottom fuel test element FTE-13 at temperatures up to 1440 °C, but no postirradiation heating tests were performed (Ref. 2).

Prior US data taken in support of fuel performance models to predict accident behavior were primarily obtained from tests with unbonded, TRISO-coated, U and Th particles rather than with compacts, characterized by relatively small sample sizes, and with a variety of fuel types heated in dry helium (e.g., Ref. 3). There is a paucity of US data in the 1400 to 1800 °C temperature range because most measurements were made at higher temperatures characteristic of large HTGR core heatup accidents; moreover, these test specimens typically had relatively high levels of as-manufactured heavy-metal contamination and coating defects.

There is a robust German data base for the performance of TRISO-coated, HEU $(Th,U)O_2$ and LEU UO_2 particles in fuel spheres under dry heating conditions; these fuel particles have performed superbly at temperatures up to 1600 °C (e.g., Ref. 4). There are also some German data for heating at higher temperatures and in the presence of high moisture which indicate degraded performance.

First-order methods have been developed, including a joint U.S./FRG accident condition model based on both U.S. and FRG dry heating data (Ref. 3). A complete uncertainty analysis for this methodology has not been conducted, but the uncertainty in the model has been estimated to be ~12x at the 95% confidence level for the FRG fuel in the range of conditions of interest. The wet heating data have not been modeled.

1.3 Data Needed

Data are needed to determine failure of TRISO-coated TF particles in fuel compacts as indicated by the release of fission products, such as Cs-137, from irradiated fuel compacts under transient conditions characteristic of pressurized and depressurized conduction cooldown events in the DB-MHR. While it is known that fuel compact matrix and OPyC coatings will oxidize under air and moisture ingress conditions, validation of the methods used to predict the rate of corrosion is needed. Furthermore, the corrosion rate and subsequent failure of the SiC under high temperature high moisture conditions is not known for irradiated fuel particles, so data are needed to make predictions on TRISO particle failure under high moisture conditions.

The data must validate that the physical models are (a) suitable for use in design codes (e.g., SORS) covering the range of expected service conditions, (b) capable of predicting the failure of high-burnup TF fuel particles to within [10x] at 95% confidence (fuel failure to be inferred from measured fission product release and PIE)

1.4 Parameters/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications as appropriate.

The irradiated TF fuel shall be heated primarily peak core heatup conditions (i.e., 1600 °C), but some testing shall done at core-average conditions. In addition, the fuel will be heated to beyond reactor conditions to quantify performances margins (i.e., fuel will be heated at high temperature (>1600 °C) until on-line fission gas release measurements indicate a failure fraction in the range of [10^{-3} to 10^{-2}]). The test conditions are the following:

Fuel First Exposed to Normal Operating Conditions

Parameter	Value
Reference fuel	TRISO-coated TF in compacts
Fuel burnup range	[0 to 85]% FIMA
Peak fast fluence	8×10^{25} n/m^2 (E >29 fJ)
System pressure*	7.1 MPa (70 atm)
Fuel temperatures/time	[800 to 1400 C]/≤36 months
Thermal Gradient	[50 - 200 C°/cm]
Environment	He/Ne(in-pile tests)
Coolant/Impurity Levels	
Normal operation (Design):	Total oxidants [≤7Pa (1ppmv)]
H_2O	[0.07 to 0.7 Pa (0.01 to 0.1 ppmv)]
CO	[1.5 to 6 Pa (0.2 to 8 ppmv)]
CO_2	[0.2 to 1.5 Pa 0.03 to 0.2 ppmv)]
H_2	[3 to 10 Pa (0.5 to 1.5 ppmv)]
CH_4	[0.3 to1.5 Pa (0.5 to 1.5 ppmv)]
N_2	[3 Pa (0.4 ppmv)]
Expected failure fraction	[10^{-5} to 10^{-3}]

Off-Normal Conditions

Reference transients	Pressurized conduction cooldown with and without water ingress Depressurized conduction
Environment	He; He/CO/H_2; He/CO/N_2
Fuel temperatures/time	
Pressurized	[1300 °C]/up to 100 h
Depressurized	[1200 to 1600 °C] from 300 to 1000 h [1600 to 2000 °C] from 100 to 200 h
System pressure	0.1 to 7.1 MPa (1 to 70 atm)
Coolant Impurity Levels	
Pressurized conditions	[0 to TBD MPa (0 to TBD atm) H_2O]
Depressurized	[0 to 35 KPa (0 to 0.35 atm)] CO [0 to 65 KPa (0 to 0.65 atm)] N_2

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*A lower pressure may be used to satisfy DDN if adequate technical justification is provided.

2. DESIGNER'S ALTERNATIVES

Alternatives to the acquisition of the above described data are:

2.1 Utilize existing accident-condition fuel performance data, primarily from heating tests with German LEU UO_2 particles in pebbles in the 1200 to 2000 ^oC temperature range; abandon the VLPC, and add a high-pressure containment and active cooling systems to the design.

2.2 Utilize FRG accident-condition models and argue that these data are directly applicable to reference U.S. fuel; abandon the VLPC, and add a high-pressure containment and active cooling systems to the design.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Fission product release data from irradiated, TF compacts will be obtained as a function of time and temperatures simulating core conduction cooldown transients. These data will allow the design methods describing performance of fuel under conduction cooldown conditions to be validated and the uncertainty interval reduced to the required accuracy.

Alternatives 2.1 and 2.2 were not chosen because of the high probability that formal validation with explicit accident-condition data for the reference TF will be required, and that significant difficulties in safety review would result (i.e., Alternatives 2.1 and 2.2 are not considered technically viable for the licensing of a VLPC). In addition the FRG fuel performance data would suffer additional risk of not complying with NQA-1 requirements.

4. SCHEDULE REQUIREMENTS

Validated design methods reflecting the expanded data base are needed 12 months prior to completion of Final Design.

5. PRIORITY

Urgency: TBD
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H

6. CONSEQUENCES OF NONEXECUTION AND FALLBACK POSITION

The fallback position is to rely upon existing fuel performance data and models and to replace the VLPC with a high-pressure containment. Even with a high-pressure containment and active cooling systems, stringent technical specifications on core operation and on containment performance should be anticipated. While such a design might be licensable, the investment risk would be considerable, and the economic viability of this design approach is questionable.

7. REFERENCES

1. "MHTGR Plutonium Consumption Study, Phase II Final Report," GA/DOE-051-94, General Atomics, April 29, 1994.

2. Miller, C. M., and W. J. Scheffel, "Postirradiation Examination and Evaluation of Peach Bottom FTE-13," GA Document 906939, Rev. N/C, GA Technologies, November 1, 1985.

3. Goodin, D.T., "US/FRG Accident Condition Fuel Performance Models," DOE-HTGR-85107, Rev. 1, General Atomics, 1989.

4. Nabielek, H., et al., "The Performance of High-Temperature Reactor Fuel Particle at Extreme Temperatures," Nucl Tech., **84**, 62-81 (1989).

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

"AAA" Fuel Program
DRIVER FUEL PROOF TEST
DDN RTW.02.15

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

A fuel proof test of the reference Driver Fuel is needed to assure that the fuel product specification, fuel process specifications, and equipment specifications have been adequately defined and reliably produce a fuel product that performs in accordance with safety review and design claims,.

1.1 Summary of Function/Assumptions

"Control Transport from Core," Assumption: Processes are available for manufacturing high-quality fuel compacts for inclusion in prismatic fuel elements.

"Retain Radionuclides in Fuel Particles with TRISO Coatings," Assumption: Processes are available for manufacturing the coated fuel particles that maintain high integrity under irradiation and accident conditions.

1.2 Current Data Base Summary

A proof test (F-30) of the TRISO-coated $(Th,U)C_2$ and ThC_2 fuel used in Fort St. Vrain was performed and served as a reliable indicator of the subsequent good performance of the fuel mass produced for the FSV reactor (Ref. 1). A formal proof test of German TRISO-coated, LEU UO_2 fuel was performed in HFR Petten in irradiation test HFR K5 and K6; no in-pile failure was observed in either test (Ref. 2). In addition, TRISO-coated, LEU UO_2 has been produced by HOBEG in significant quantities for AVR reloads (~6000 fuel spheres) which effectively served as large-scale proof tests; an initial reload performed relatively poorly, but subsequent reloads performed superbly.

[An initial core of high-quality of LEU UO_2 fuel was produced by NFI for the Japanese 30 MW(t) HTTR. JAERI conducted an extensive irradiation program in support of fuel manufacturing for the HTTR; whether a formal proof test was performed is unknown at this writing.]

1.3 Data Needed

Data [fuel failure fraction as inferred from fission gas release (Kr and Xe) release and fission metal release (Cs-137)] are needed to confirm that Driver Fuel from the final demonstration line which meets the DB-MHR Fuel Product Specification and has been manufactured in accordance with the DB-MHR Fuel Process Specification and Equipment Specifications exhibits a mean observed failure at 95% confidence within [4x] of that predicted by the fuel performance models (models which will be constructed primarily from test data obtained for developmental fuels, some of which were fabricated with laboratory-scale equipment).

Since the fuel coatings provide the primary barrier to fission product release, proof test data are needed to show that the optimized, final demonstration line processes deliver Driver Fuel with the required performance. The proof test of Driver Fuel shall consist of: (1) irradiation of a statistically significant quantity of DF compacts to bounding service conditions (burnup, fast fluence and temperature), (2) a comprehensive postirradiation examination, (3) postirradiation heating to at least 1600 °C and (4) documentation of test results.

1.4 Parameters/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications as appropriate. The proof test Driver Fuel will be irradiated to normal peak conditions and heated at peak core heatup conditions (1600 °C).

Proof Test Driver Fuel First Exposed to Bounding Normal Service Conditions

Parameter	Value
Reference fuel	TRISO-coated Driver Fuel in compacts
Fuel burnup range	[0 to 85]% FIMA
Peak fast fluence	8 x 10^{25} n/m^2 (E >29 fJ)
System pressure[*]	7.1 MPa (70 atm)
Fuel temperatures/time	[≥1250 °C]/≤36 months
Environment	He/Ne(in-pile tests)
Coolant/Impurity Levels	
Normal operation (Design):	Total oxidants [≤7Pa (1ppmv)]
H_2O	[0.07 to 0.7 Pa (0.01 to 0.1 ppmv)]
CO	[1.5 to 6 Pa (0.2 to 8 ppmv)]
CO_2	[0.2 to 1.5 Pa 0.03 to 0.2 ppmv)]
H_2	[3 to 10 Pa (0.5 to 1.5 ppmv)]
CH_4	[0.3 to1.5 Pa (0.5 to 1.5 ppmv)]
N_2	[3 Pa (0.4 ppmv)]
Expected failure fraction (50% Confidence)	[≤1 x 10^{-4}]
Off-Normal Conditions	
Environment	He; He/CO/H_2; He/CO/N_2
Fuel temperatures/time	[1600 °C for 100 hr]
System pressure	0.1 to 7.1 MPa (1 to 70 atm)

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[*] A lower pressure may be used to satisfy DDN if adequate technical justification is provided.

Coolant Impurity Levels	[0 to 35 KPa (0 to 0.35 atm) CO] [0 to 65 KPa (0 to 0.65 atm) N_2]
Expected failure fraction (50% Confidence)	[TBD]

2. DESIGNER'S ALTERNATIVES

Alternatives to the acquisition of the above described data are:

2.1 Utilize performance data from the initial DF demonstration line capsule tests; correlate the as-manufactured attributes of this fuel with that of the final DF bench-scale fuel; and assume that these as-manufactured fuel attributes, as determined by standard QC techniques, are reliable indicators of inservice fuel performance. In other words, assume that any performance differences in the two fuels can be reliably determined in advance by characterizing the as-manufactured product.

2.2 Argue that favorable U.S. and FRG experience with the mass-production of TRISO-coated HEU and LEU fuels, including FSV fuel, demonstrates that proof tests are unnecessary, i.e., that process and equipment scale up do not impact significantly inservice performance of TRISO-coated fuel particles.

3. SELECTED APPROACH AND EXPLANATION

Obtain irradiation performance data for DF compacts made on a fully developed production line in accordance with the DB-MHR Fuel Product, Process, and Equipment Specifications. Alternative 2.1 leaves too much risk of nonconformance with in-reactor fuel performance requirements since highly reliable correlations between as-manufactured fuel attributes and inservice performance have not yet been developed and validated. Alternative 2.2 is not credible, given the history of serious difficulties and frequent surprises in the scale up of processes and equipment for the manufacture of nuclear reactor fuels.

4. SCHEDULE REQUIREMENTS

[The test must be completed and documented prior to operation of the initial core.]

5. PRIORITY

Urgency: TBD
Cost benefit: H
Uncertainty in existing data: M
Importance of new data: H

6. CONSEQUENCES OF NONEXECUTION AND FALLBACK POSITION

The fallback position is to utilize data from the initial demonstration line capsule test based on fuel from an incompletely developed demonstration line. The consequences are expected to be (1) difficulties and delays in safety review while technical arguments of the similarities between initial and final fuel are debated; (2) increased risk that the initial core will not perform as well as predicted because of unanticipated differences in fuel from the completed process line; (3) stringent technical specifications on core operation until the initial core reaches full-burnup; and (4) excessive reactor surveillance requirements, including multiple destructive fuel-element PIEs.

7. REFERENCES

1. Scott, C.B, and D.P. Harmon, “Postirradiation Examination of Capsule F-30,” GA-A13208, General Atomic, April 1975.
2. “Fuel Performance and Fission Product Behavior in Gas Cooled Reactors,” IAEA-TECDOC-978, International Atomic Energy Agency, November 1997, p. 68.

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

“AAA” Fuel Program
TRANSMUTATION FUEL PROOF TEST
DDN RTW.02.16

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

A fuel proof test of the reference Transmutation Fuel is needed to assure that the fuel product specification, fuel process specifications, and equipment specifications have been adequately defined and reliably produce a fuel product that performs in accordance with safety review and design claims,.

1.1 Summary of Function/Assumptions

"Control Transport from Core," Assumption: Processes are available for manufacturing high-quality fuel compacts for inclusion in prismatic fuel elements.

"Retain Radionuclides in Fuel Particles with TRISO Coatings," Assumption: Processes are available for manufacturing the coated fuel particles that maintain high integrity under irradiation and accident conditions.

1.2 Current Data Base Summary

A proof test (F-30) of the TRISO-coated $(Th,U)C_2$ and ThC_2 fuel used in Fort St. Vrain was performed and served as a reliable indicator of the subsequent good performance of the fuel mass produced for the FSV reactor (Ref. 1). A formal proof test of German TRISO-coated, LEU UO_2 fuel was performed in HFR Petten in irradiation test HFR K5 and K6; no in-pile failure was observed in either test (Ref. 2). In addition, TRISO-coated, LEU UO_2 has been produced by HOBEG in significant quantities for AVR reloads (~6000 fuel spheres) which effectively served as large-scale proof tests; an initial reload performed relatively poorly, but subsequent reloads performed superbly.

[An initial core of high-quality of LEU UO_2 fuel was produced by NFI for the Japanese 30 MW(t) HTTR. JAERI conducted an extensive irradiation program in support of fuel manufacturing for the HTTR; whether a formal proof test was performed is unknown at this writing.]

1.3 Data Needed

Data [fuel failure fraction as inferred from fission gas release (Kr and Xe) release and fission metal release (Cs-137)] are needed to confirm that Transmutation Fuel from the final demonstration line which meets the DB-MHR Fuel Product Specification and has been manufactured in accordance with the DB-MHR Fuel Process Specification and Equipment Specifications exhibits mean observed failure at 95% confidence within [10x] of that predicted by the fuel performance models (models which will be constructed primarily from test data obtained for developmental fuels, some of which were fabricated with laboratory-scale equipment).

Since the fuel coatings provide the primary barrier to fission product release, proof test data are needed to show that the optimized, final TF production line processes deliver TF with the required performance. The proof test of Transmutation Fuel shall consist of: (1) irradiation of a statistically significant quantity of TF compacts to bounding service conditions (burnup, fast fluence and temperature), (2) a comprehensive postirradiation examination, (3) postirradiation heating to at least 1600 °C, and (4) documentation of test results.

1.4 Parameters/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications as appropriate. The proof test Transmutation Fuel will be irradiated to normal peak conditions and heated at peak core heatup conditions (1600 °C).

Proof Test Transmutation Fuel First Exposed to Bounding Normal Conditions

Parameter	Value
Reference fuel	TRISO-coated Transmutation Fuel in compacts
Fuel burnup range	[0 to 85]% FIMA
Peak fast fluence	8×10^{25} n/m^2 (E >29 fJ)
System pressure*	7.1 MPa (70 atm)
Fuel temperatures/time	[≥1250 °C]/≤36 months
Environment	He/Ne(in-pile tests)
Coolant/Impurity Levels	
Normal operation (Design):	Total oxidants [≤7Pa (1ppmv)]
H_2O	[0.07 to 0.7 Pa (0.01 to 0.1 ppmv)]
CO	[1.5 to 6 Pa (0.2 to 8 ppmv)]
CO_2	[0.2 to 1.5 Pa 0.03 to 0.2 ppmv)]
H_2	[3 to 10 Pa (0.5 to 1.5 ppmv)]
CH_4	[0.3 to1.5 Pa (0.5 to 1.5 ppmv)]
N_2	[3 Pa (0.4 ppmv)]
Expected failure fraction (50% Confidence)	[≤1 x 10^{-4}]
Off-Normal Conditions	
Environment	He; He/CO/H_2; He/CO/N_2
Fuel temperatures/time	[1600 °C for 100 hr]

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*A lower pressure may be used to satisfy DDN if adequate technical justification is provided.

System pressure	0.1 to 7.1 MPa (1 to 70 atm)
Coolant Impurity Levels	[0 to 35 KPa (0 to 0.35 atm) CO] [0 to 65 KPa (0 to 0.65 atm) N_2]
Expected failure fraction (50% Confidence)	[TBD]

2. DESIGNER'S ALTERNATIVES

Alternatives to the acquisition of the above described data are:

2.1 Utilize performance data from the initial TF demonstration line capsule tests; correlate the as-manufactured attributes of this fuel with that of the final TF bench-scale fuel; and assume that these as-manufactured fuel attributes, as determined by standard QC techniques, are reliable indicators of inservice fuel performance. In other words, assume that any performance differences in the two fuels can be reliably determined in advance by characterizing the as-manufactured product.

2.2 Argue that favorable U.S. and FRG experience with the mass-production of TRISO-coated HEU and LEU fuels, including FSV fuel, demonstrates that proof tests are unnecessary, i.e., that process and equipment scale up do not impact significantly inservice performance of TRISO-coated fuel particles.

3. SELECTED APPROACH AND EXPLANATION

Obtain irradiation performance data for TF compacts made on a fully developed production line in accordance with the DB-MHR Fuel Product, Process, and Equipment Specifications. Alternative 2.1 leaves too much risk of nonconformance with in-reactor fuel performance requirements since highly reliable correlations between as-manufactured fuel attributes and inservice performance have not yet been developed and validated. Alternative 2.2 is not credible, given the history of serious difficulties and frequent surprises in the scale up of processes and equipment for the manufacture of nuclear reactor fuels.

4. SCHEDULE REQUIREMENTS

[The test must be completed and documented prior to operation of the initial core.]

5. PRIORITY

Urgency: TBD
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H

6. CONSEQUENCES OF NONEXECUTION AND FALLBACK POSITION

The fallback position is to utilize data from the initial demonstration line capsule test based on fuel from an incompletely developed demonstration line. The consequences are expected to be (1) difficulties and delays in safety review while technical arguments of the similarities between initial and final fuel are debated; (2) increased risk that the initial core will not perform as well as predicted because of unanticipated differences in fuel from the completed process line; (3) stringent technical specifications on core operation until the initial core reaches full-burnup; and (4) excessive reactor surveillance requirements, including multiple destructive fuel-element PIEs.

7. REFERENCES

1. Scott, C.B, and D.P. Harmon, "Postirradiation Examination of Capsule F-30," GA-A13208, General Atomic, April 1975.

2. "Fuel Performance and Fission Product Behavior in Gas Cooled Reactors," IAEA-TECDOC-978, International Atomic Energy Agency, November 1997, p. 68.

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Originator Date

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Engineering Manager Date

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Project Manager Date

DATE: 09/27/02

"AAA" Fuel Program
FISSION GAS RELEASE FROM CORE MATERIALS
DDN RTW.03.01

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The dominant sources of fission gas release, including iodine and tellurium isotopes, are heavy-metal contamination in the fuel-compact matrix and failed fuel particles with exposed kernels; consequently, the release characteristics of these two sources must be determined for both DF and TF compacts, including the effects of environmental and irradiation conditions, for normal operating conditions and for dry and wet core conduction cooldown conditions.

1.1 Summary of Function/Assumptions

"Retain Radionuclides in Fuel Particles," Assumption: Reference correlations for fission gas release from heavy metal contamination are accurate to within [<4x] at 95% confidence.

"Retain Radionuclides in Fuel Kernels," Assumption: Exposed fuel kernels will hydrolyze during irradiation.

"Retain Radionuclides in Fuel Kernels," Assumption: Reference correlations for transport of fission products in fuel kernels are accurate to within [<4x] at 95% confidence.

"Protect the Capability to Retain Radionuclides in Fuel Kernels," Assumption: Reference correlations describe fission product release from kernels under dry and wet core conduction cooldown conditions to within a factor of [4x] at 95% confidence.

"Retain Radionuclides in Fuel Particles," Assumption: No incremental release from heavy metal contamination as a result of steam ingress.

"Retain Radionuclides in Fuel Kernels," Assumption: Reference correlations describe fission product release from kernels under dry and wet core conduction cooldown conditions to within a factor of [<4x] at 95% confidence.

1.2 Current Data Base Summary

The present data base for fission gas release from heavy-metal contamination and from failed particles is derived primarily from TRIGA measurements on fuel compact matrix doped with uranium and on laser-failed, irradiated fuel particles, respectively (e.g., Ref. 1) . The effects of fuel hydrolysis (reaction of exposed kernels with water) on gas release are derived largely from laboratory measurements and short-term TRIGA tests. Isothermal, in-pile hydrolysis tests on LEU UCO fuel (HRB 17/18) were completed at ORNL, and the temperature dependence of gas release from both unhydrolyzed and hydrolyzed fuel has been addressed in the HFR B1 test in Petten (Ref. 2).

The HFR B1 data have been used to derive a comprehensive new model for fission gas release from unhydrolyzed exposed LEU UCO kernels. These test data indicate no strong burnup dependence for fission gas release from LEU UCO kernels up to a burnup of ~18% FIMA. Circumstantial evidence from the NPR-1/-2 irradiations of HEU UCO fuel to ~75% FIMA suggest a large burnup dependence (5-10x increases) at the higher burnups. Limited German data for high-burnup UO_2 also suggest a burnup dependence.

The present data base for fission gas release from failed particles under depressurized core conduction conditions is derived largely from measurements on laser-failed HEU UC_2/ThO_2 particles; the US iodine release data are exclusively from this source. In addition, ~18% FIMA LEU UCO particles were recovered from the HFR B1 test, mechanically failed and heated at KFA; 50-100% of the Kr-85 was rapidly released upon heating to only 800 ^{o}C (Ref. 3). Integral Kr, Xe and I release data for 8-10% FIMA LEU UO_2 fuel spheres are also available from the German fuel heating program.

There are no direct measurements of the fission gas release characteristics of coated-particle Pu fuels.

1.3 Data Needed

Measurements are needed of the fission gas release rates (Kr, Xe, I, and Te) from heavy-metal contamination and from failed reference DF and TF particles as a function of kernel composition, temperature, half-life, burnup and flux under irradiation and under dry and wet core conduction cooldown conditions. In addition, the effect of hydrolysis on gas release must be quantified for steady-state irradiation and for transient wet core conduction cooldown conditions. The assumption that I and Te isotopes behave like Xe isotopes under irradiation also must be confirmed. The releases of I-131 from exposed DF and TF kernels must be measured directly under core conduction cooldown conditions. Sufficient data are needed to develop and refine gas release models with uncertainties ≤4x at 95% confidence.

1.4 Parameters/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications:

Parameter	Value
Reference fuel	TRISO-coated DF and TF in compacts
DF burnup range	[0 to 85]% FIMA
TF burnup range	[0 to 85]% FIMA
DF Peak fast fluence	[8 x 10^{25} n/m^2 (E >29 fJ)]
TF Peak fast fluence	[8 x 10^{25} n/m^2 (E >29 fJ)]
System pressure[*]	
Normal condition	7.1 MPa (70 atm)
Off-normal conditions	0.1 to 7.1 MPa (1 to 70 atm)
Fuel temperatures/time	
Normal condition	[800 to 1400 C]/≤36 months
Off-normal conditions (core conduction cooldown)	
Pressurized (wet)	[≤1400 °C]/≤100 h
Depressurized	[1200 to 2000 °C] /100 to 1000h
Environment	
Normal conditions	He/Ne(in-pile tests)
Off-normal conditions	He; He/H_2O/CO/H_2; He/CO/N_2
Coolant/Impurity Levels	
Normal condition (Design):	Total oxidants [≤7Pa (1ppmv)]
H_2O	[0.07 to 0.7 Pa (0.01 to 0.1 ppmv)]
CO	[1.5 to 6 Pa (0.2 to 8 ppmv)]
CO_2	[0.2 to 1.5 Pa 0.03 to 0.2 ppmv)]
H_2	[3 to 10 Pa (0.5 to 1.5 ppmv)]
CH_4	[0.3 to1.5 Pa (0.5 to 1.5 ppmv)]
N_2	[3 Pa (0.4 ppmv)]
Pressurized condition	
H_2O	[0 to TBD KPa (0 to TBD atm)]
Depressurized condition	
CO	[0 to 35 KPa (0 to 0.35 atm)]
N_2	[0 to 65 KPa (0 to 0.65 atm)]
Fission gases of interest	I, Te > Kr > Xe[**]

2. DESIGNER'S ALTERNATIVES

The following alternative has been considered:

2.1 Assume 100% release of Kr, Xe, I and Te from failed DF and TF particles during normal operation and postulated accidents.

2.2 Use existing HEU UC_2/ThO_2 data bases and models for DF and TF compacts.

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[*] A lower pressure may be used to satisfy DDN if adequate technical justification is provided.

[**] Radionuclides ordered according to radiological significance.

2.3 Use German 10%-enriched UO_2 data base and models, recognizing that these low-burnup data likely underestimate the gas releases from higher-burnup, DF and TF kernels; hence, the effects of burnup would have to be estimated from other data.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Measure fission gas release (Kr, Xe, I, and Te) from exposed DF and TF kernels irradiated under near normal DB-MHR flux over a range of temperatures. Measure the fission gas, including I-131, from irradiated reference DF and TF compacts under dry and wet core conduction cooldown conditions. Such measurements will reduce the uncertainties in the fission gas retention characteristics of the reference fuel and provide a basis for judging the adequacy of the present design.

Alternative 2.1 would grossly overestimate the release of Kr, Xe, I and Te isotopes from the core during normal operation and postulated accidents; assumption of 100% I-131 release during core conduction cooldown conditions would preclude the use of the VLPC for the DB-MHR. Alternative 2.2 can not be justified technically because of the absence of accident-condition data for DF and TF kernels and the expected differences between DF and TF and HEU UC_2/ThO_2 fuel. Alternative 2.3 can not be used directly because of the absence of sufficient burnup data and potentially large burnup and kernel composition effects; however, the effect of burnup may be estimated from experimental data for other kernel compositions.

The recommended test program will consist of: (1) R/B, hydrolysis, and gas release experiments performed on laser-failed, reference DF and TF particles; (2) reactivation and heating of laser-failed, irradiated DF and TF particles; and (3) irradiation and postirradiation heating of fuel compacts containing known fractions of "designed-to-failed" DF and TF particles (reference kernels with a 10-15 μm pyrocarbon seal coat).

4. SCHEDULE REQUIREMENTS

Preliminary data [6] months before start of Preliminary Design and final data [12] months prior to completion of Final Design.

5. PRIORITY

Urgency: TBD
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H

6. CONSEQUENCES OF NONEXECUTION AND FALLBACK POSITION

Assume 100% release from contamination and failed particles during normal operation (alternative 2.1), and use the best available FRG data for release from oxide fuels under core conduction cooldown conditions with a safety margin added for anticipated burnup effects (alternative 2.3). Accept the attendant large conservatism in the releases of Kr, Xe, I and Te isotopes from the core during normal operation and the uncertainty in I release during accidents. The results would be unfounded O&M concerns, high projected occupational exposures, and the imposition of onerous technical specifications on PCS circulating and plateout inventories. A bounding assumption of 100% release for exposed DF and TF kernels in the DB-MHR with a VLPC can not be tolerated for compliance with PAGs. In other words, assuming 100% I-131 release precludes the use of the VLPC on the DB-MHR

7. REFERENCES

1. "Fuel Performance and Fission Product Behavior in Gas Cooled Reactors," IAEA-TECDOC-978, International Atomic Energy Agency, November 1997, p. 217-257.

2. Myers, B.F., "Effect of Water Vapor on the Release of Fission Gases from Uranium Oxycarbide in High-Temperature, Gas-Cooled Reactor Coated Fuel Particles," J. Am Ceram. Soc. 75, 686-693 (1992).

3. Myers, B.F., "The Effect of Water Vapor on the Release of Fission Gas from the Fuel Elements of High-Temperature, Gas-Cooled Reactors: A Preliminary Assessment of Experiments HRB-17, HFR-B1, HFR-K6 and KORA," ORNL/M-4294, ORNL, 1995.

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

“AAA” Fuel Program
FISSION METAL DIFFUSIVITIES IN FUEL KERNELS
DDN RTW.03.02

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

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 the reference DF/TF kernels must be characterized for normal operating conditions and for core conduction cooldown transients.

1.1 Summary of Function/Assumptions

"Retain Radionuclides in Fuel Kernels," Assumption: Reference correlations for fission metal transport in fuel kernels are accurate to within a factor of [<10x] at 95% confidence.

"Protect the Capability to Retain Radionuclides in Fuel Kernels," Assumption: Reference correlations for fission metal transport under core conduction cooldown conditions are accurate to within a factor of [<10x] at 95% confidence.

1.2 Current Data Base Summary

The present data base is derived primarily from measurement on particles irradiated in accelerated test capsules (e.g., Refs. 1 and 2). There are FRG data for Cs, Sr, and Ag in oxide kernels of intact particles which were irradiated under near real-time conditions as well as limited laboratory data on Cs release from ThO_2 kernels. A large number piggy-back samples of bare kernels and coated particles with various kernel compositions, including LEU UCO, have been irradiated in numerous irradiation capsules, but these piggy-back samples have never been systematically evaluated.

The reference correlations for Cs, Sr and Ag diffusivities in LEU UCO kernels are characterized by very large burnup dependencies (Ref. 3) which need to be confirmed experimentally.

There are no kernel release data for PuO_x kernels or for the kernel compositions used in the reference DF and TF particles.

1.3 Data Needed

Correlations are needed for the release and the attendant effective diffusivities of key fission metals (Cs, Ag and Sr) for DF/TF kernels as a function of temperature, burnup and, if appropriate, neutron flux for normal operation and dry and wet core conduction cooldown conditions. The tentative observation that the metal diffusivities in the kernels of intact particles are significantly lower than in the kernels of failed particles also needs to be confirmed and quantified. Sufficient data are needed to develop and refine diffusivity correlations with uncertainties [<10x] at 95% confidence level.

1.4 Parameters/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications as appropriate:

Parameter	Value
Test Articles	TRISO-coated DF/TF kernels
DF burnup range	[0 to 85]% FIMA
TF burnup range	[0 to 85]% FIMA
DF Peak fast fluence	[8 x 10^{25} n/m^2 (E >29 fJ)]
TF Peak fast fluence	[8 x 10^{25} n/m^2 (E >29 fJ)]
System pressure[*]	
Normal condition	7.1 MPa (70 atm)
Off-normal conditions	0.1 to 7.1 MPa (1 to 70 atm)
Fuel temperatures/time	
Normal condition	[800 to 1400 C]/≤36 months
Off-normal conditions (core conduction cooldown)	
Pressurized (wet)	[≤1400 °C]/≤100 h
Depressurized	[1200 to 2000 °C] /100 to 1000h
Environment	
Normal conditions	He/Ne(in-pile tests)
Off-normal conditions	He; He/H_2O/CO/H_2; He/CO/N_2
Coolant/Impurity Levels	
Normal condition (Design):	Total oxidants [≤7Pa (1ppmv)]
H_2O	[0.07 to 0.7 Pa (0.01 to 0.1 ppmv)]
CO	[1.5 to 6 Pa (0.2 to 8 ppmv)]
CO_2	[0.2 to 1.5 Pa 0.03 to 0.2 ppmv)]
H_2	[3 to 10 Pa (0.5 to 1.5 ppmv)]
CH_4	[0.3 to1.5 Pa (0.5 to 1.5 ppmv)]
N_2	[3 Pa (0.4 ppmv)]
Pressurized condition	
H_2O	[0 to TBD KPa (0 to TBD atm)]

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[*] A lower pressure may be used to satisfy DDN if adequate technical justification is provided.

Depressurized condition	
CO	[0 to 35 KPa (0 to 0.35 atm)]
N_2	[0 to 65 KPa (0 to 0.65 atm)]
Fission metals of interest	Cs, Ag > Sr

2. DESIGNER'S ALTERNATIVES

The following alternatives have been considered:

2.1 Take no credit for kernel retention when calculating fission metal release rates from the core.

2.2 Use current LEU UCO/ThO_2 models which hypothesize large burnup dependencies and assume that irradiation conditions, particularly high neutron fluxes and high temperatures, have no special effects on fission metal release from fuel kernels.

2.3 Use FRG correlations for UO_2 kernel diffusivities with an added burnup dependence, based upon data for other kernel compositions.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Measure fission metal release from DF/TF fuel kernels in failed and intact particles under near real-time irradiation and core conduction cooldown conditions. The estimated uncertainties in the reference correlations are excessively large; one major source of uncertainty is that these correlations are based largely on data from accelerated irradiation tests which may significantly overestimate kernel release under real-time conditions.

Alternative 2.1 would grossly overestimate the release of Sr and Cs isotopes from the core during normal operation and postulated accidents (degree of conservatism for Ag isotopes may be modest). Alternative 2.2 can not be justified technically because of potential differences between high-burnup DF/TF kernels and much lower-burnup LEU UCO/ThO_2 kernels. Alternative 2.3 can not be justified technically because of potentially large burnup effects.

The recommended test program will utilize “designed-to-fail” DF/TF fuel particles (reference kernels with a 10-15 μm pyrocarbon seal coat) irradiated in piggy-back samples and in seeded fuel compacts. Data will be obtained by examination of fuel particles from irradiation capsules and from postirradiation heating of selected samples.

4. SCHEDULE REQUIREMENTS

Preliminary data [6] months before start of Preliminary Design and final data [12] months prior to completion of Final Design.

5. PRIORITY

Urgency: TBD
Cost benefit: M
Uncertainty in existing data: H
Importance of new data: M

6. CONSEQUENCES OF NONEXECUTION AND FALLBACK POSITION

Alternative 2.1 along with a more conservative plant design to account for not taking credit for retention of fission metals in fuel kernels and accept (grossly?) conservative estimates of the releases of Cs, Ag, and Sr isotopes from the core during normal operation and postulated accidents, resulting in increased O&M concerns and, ultimately, D&D concerns, high projected occupational exposures, and the imposition of onerous technical specifications on containment building performance.

Failure to fully exploit the inherent retentivity of oxide-based TRISO particles, especially at low- to moderate burnups, will necessitate more reliance upon the core graphite as a barrier to release of fission metals.

7. REFERENCES

1. "Fuel Performance and Fission Product Behavior in Gas Cooled Reactors," IAEA-TECDOC-978, International Atomic Energy Agency, November 1997, p. 436-437.

2. Myers, B.F., "Kernel Diffusion Coefficients," GA Document 906031, Rev. 1, General Atomic, July 1981.

3. Myers, B.F., "Fuel Design Data Manual," GA Document 901866, Ref. F, General Atomics, August 1987.

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

“AAA” Fuel Program
ACTINIDE RELEASE FROM FUEL KERNELS
DDN RTW.03.03

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The fuel kernel of the coated particle is the initial barrier to the release of radionuclides from the core and should provide significant, if not complete, retention of the transuranics even under core heatup conditions (the exception may be americium); consequently, the transport properties of Minor Actinides in the reference DF and TF kernels must be characterized for normal operating conditions and for core conduction cooldown transients. The rate limiting step in the potential release of Minor Actinides from the DF and TF kernels may be vaporization from the kernel exterior surface and/or diffusion in the kernel itself. In a predominantly Pu kernel (i.e., DF and TF kernels), Pu diffusion would play a much smaller role than for kernels in which the Pu is a minor constituent (e.g., ThO_2, UCO, UC_2, and UO_2).

1.1 Summary of Function/Assumptions

"Retain Radionuclides in Fuel Kernels," Assumption: Reference correlations for fission metal transport in fuel kernels are accurate to within a factor of [<10x] at 95% confidence.

"Protect the Capability to Retain Radionuclides in Fuel Kernels," Assumption: Reference correlations for fission metal transport under core conduction cooldown conditions are accurate to within a factor of [<10x] at 95% confidence.

1.2 Current Data Base Summary

The releases of plutonium, americium and curium from kernels of various compositions [$(Th,U)O_2$, UO_2, UC_2 and UCO] under irradiation at high temperature (1100 – 1600 ^{o}C) have been investigated (e.g., Refs. 1, 2 and 3). The fractional releases from the kernels to the IPyC layer in intact particles were strongly dependent kernel composition: the actinides appear to be completely retained by the UO_2 kernel, but some release was observed from kernels containing as little as 3% UC_2; the apparent diffusion coefficient for Am in UC_2 was an order of magnitude higher than that for Pu at 1350 ^{o}C (Ref. 1).

The diffusivity of Pu in irradiated MOX pellets [$(U,Pu)O_{2-x}$] has been measured in the 1600 – 2500 °C range and shown to be a function of temperature, burnup and kernel stoichiometry (e.g., Ref. 4). However, the actinides in MOX fuel are typically fully saturated with oxygen; substoichiometric DF and TF kernels may behave differently.

There are no available data on the release of actinides from failed TRISO-coated PuO_x particles.

1.3 Data Needed

Correlations are needed for the release of Pu and Np isotopes from DF kernels and of Pu isotopes and key Minor Actinides (Am, Cm and Np) from TF kernels as a function of kernel composition, temperature, burnup and, if appropriate, neutron flux, for normal operation and dry and wet core conduction cooldown conditions. The tentative observation that the actinide releases from the kernels of intact particles are significantly lower than from the kernels of failed particles also needs to be confirmed and quantified. Sufficient data are needed to develop and refine diffusivity correlations with uncertainties [<10x] at 95% confidence level.

1.4 Parameters/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications as appropriate:

Parameter	Value
Test articles	TRISO-coated DF/TF kernels
DF burnup range	[0 to 85]% FIMA
TF burnup range	[0 to 85]% FIMA
DF Peak fast fluence	[8 x 10^{25} n/m^2 (E >29 fJ)]
TF Peak fast fluence	[8 x 10^{25} n/m^2 (E >29 fJ)]
System pressure*	
Normal condition	7.1 MPa (70 atm)
Off-normal conditions	0.1 to 7.1 MPa (1 to 70 atm)
Fuel temperatures/time	
Normal condition	[800 to 1400 C]/≤36 months
Off-normal conditions (core conduction cooldown)	
Pressurized (wet)	[≤1400 °C]/≤100 h
Depressurized	[1200 to 2000 °C] /100 to 1000h
Environment	
Normal conditions	He/Ne(in-pile tests)
Off-normal conditions	He; He/H_2O/CO/H_2; He/CO/N_2
Coolant/Impurity Levels	
Normal condition (Design):	Total oxidants [≤7Pa (1ppmv)]

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*A lower pressure may be used to satisfy DDN if adequate technical justification is provided.

H_2O	[0.07 to 0.7 Pa (0.01 to 0.1 ppmv)]
CO	[1.5 to 6 Pa (0.2 to 8 ppmv)]
CO_2	[0.2 to 1.5 Pa 0.03 to 0.2 ppmv)]
H_2	[3 to 10 Pa (0.5 to 1.5 ppmv)]
CH_4	[0.3 to1.5 Pa (0.5 to 1.5 ppmv)]
N_2	[3 Pa (0.4 ppmv)]
Pressurized condition	
H_2O	[0 to TBD KPa (0 to TBD atm)]
Depressurized condition	
CO	[0 to 35 KPa (0 to 0.35 atm)]
N_2	[0 to 65 KPa (0 to 0.65 atm)]
Radionuclides of interest	Am > Pu > Cm, Np

2. DESIGNER'S ALTERNATIVES

The following alternatives have been considered:

2-1. Take no credit for kernel retention when calculating actinide release rates from the core; limit the coating failure fraction as necessary, and take credit for retention by the fuel-element graphite.

2-2. Use the limited data of Ref. 3 for UCO which indicate 1-10% kernel release of Pu and Am in intact TRISO-coated particles.

2-3. Estimate vapor pressures of Pu, Np, Am and Cm from chemical thermodynamic data, and use these vapor pressures to estimate retention in the kernel.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Measure actinide release from DF/TF kernels in failed and intact particles under near real-time irradiation conditions and core conduction cooldown conditions. There are no available data that can be credibly used to estimated the release rates of actinides from high-burnup DF and TF kernels. Alternatives 2.1 and 2.2 would grossly overestimate the release of Pu isotopes and Minor Actinides from failed particles during normal operation and postulated accidents. Alternative 2.3 is compromised by excessively large uncertainties when applied to high-burnup DF and TF particles.

The recommended test program will utilize “designed-to-fail” DF/TF particles (reference kernels with a 10-15 µm pyrocarbon seal coat) irradiated in piggy-back samples and in seeded fuel compacts. Data will be obtained by examination of fuel particles from irradiation capsules and from postirradiation heating of selected samples.

4. SCHEDULE REQUIREMENTS

Preliminary data [6] months before start of Preliminary Design and final data [12] months prior to completion of Final Design.

5. PRIORITY

Urgency: TBD
Cost benefit: M
Uncertainty in existing data: H
Importance of new data: M

6. CONSEQUENCES OF NONEXECUTION AND FALLBACK POSITION

Alternative 1 along with a more conservative plant design to account for not taking credit for retention of actinides in DF/TF kernels and rely upon low coating failure limits and the matrix and graphite to limit their release from the core during normal operation and postulated accidents, resulting in increased O&M concerns and, ultimately, D&D concerns, high projected occupational exposures, and the imposition of onerous technical specifications on containment building performance.

7. REFERENCES

1. Foerthmann, R., and E. Vekens, "Transportverhalten von Plutonium and Americium in niedrig angereicherten Beschichteten Brennstoffteilchen bei hohen Bestrahlungstemperaturen," Juel-1811, KFA, Oktober, 1982.

2. e. Silva, A.T., "Experimentalle Untersuchungen der Aktinidenfreisetzung aus Brennstoffteilchen fuer Hochtemperaturreaktoren," Juel-1833, Maerz 1983.

3. Mehner, A.W., et al., "Performance of Ceramic HTR Fuel and Coating Materials under Normal and Transient Conditions," Proceedings of *Gas-Cooled Reactors Today*, Vol. 2, BNES, pp. 139-144, 1982.

4. Chilton, C.R., and J. Edwards, "The Solid-state Diffusion of Plutonium in Uranium Oxide," J. Nucl. Mater., **78**, 182-191 (1978).

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

“AAA” Fuel Program
RADIONUCLIDE DIFFUSIVITIES IN PARTICLE COATINGS
DDN RTW.03.04

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The fuel particle coatings, particularly the SiC coating, are the primary barrier to release of fission products from the core during normal operation and during core conduction cooldown transients; consequently, the effective diffusivities of fission products in particle coatings must be determined as a function of environmental and irradiation conditions.

1.1 Summary of Function/Assumptions

"Retain Radionuclides with Particle Coatings," Assumption: Reference correlations are adequate to describe fission product transport in SiC and PyC coatings to within factor of [<10x] at 95% confidence.

"Protect the Capability to Retain Radionuclides with Particle Coatings," Assumption: Reference correlations for radionuclide transport in particle coatings are accurate to within [<10x] at 95% confidence.

"Retain Radionuclides with Particle Coatings," Assumption: Reference correlations for radionuclide transport in particle coatings under core conduction cooldown conditions are accurate to within [<10x] at 95% confidence.

1.2 Current Data Base Summary

The reference diffusivity correlations for various fission products in SiC and pyrocarbon coatings were largely calculated from particle release measurements in a laboratory environment (e.g., Ref. 1). These data are supported by limited in-pile data for Cs, Sr and Ag calculated from the results of irradiation experiments. There are limited and highly variable data on the diffusive release of fission gases from BISO particles, but the relevance of these data to the transport of gases in the OPyC coatings of TRISO particles is questionable.

The key issue is the diffusive release of silver through the SiC of intact TRISO fuel particles at high temperatures. There are data from irradiation capsules, operating reactors and postirradiation heating tests that clearly demonstrate that a significant fraction (>1%) of the silver can be released from intact TRISO particles; however, there is considerable scatter in the Ag-in-SiC diffusion coefficients derived from these data (e.g., Ref. 1).

One complication is that undetected SiC defects in the test sample will bias high the apparent Ag diffusion coefficient calculated from the integral fractional release. A further complication is that large particle-to-particle variations in the Ag fractional releases are observed for populations that are ostensibly free of SiC defects based upon their retention of other volatile fission metals (IMGA at ORNL, Ref. 2). No satisfactory explanation has been given to date.

There are limited data regarding the diffusion of Pu, Am and Cm in PyC and SiC coatings on coated, uranium fuel particles (e.g., Refs. 3 - 5). The diffusion of these actinides in HTI and LTI PyC appeared to be relatively rapid at high temperature. The actinides were quantitatively retained by SiC to at least 1400 °C; measurable releases were reported at 1600 °C, but the condition of the SiC coatings in these experiments is unknown (Ref. 4). There are no data available for Pu release from failed PuO_x particles or transport in PyC and SiC coatings on coated, Pu fuel particles.

1.3 Data Needed

The effective diffusivities of key radionuclides in the coatings of DF and TF particles are needed as a function of temperature and, as required, of fluence, irradiation temperature, and as-manufactured coating attributes for normal operation and for core conduction cooldown conditions; specifically, the effective diffusivities of the volatile fission metals (Ag, Cs, and Sr) in SiC coatings are needed as are the diffusivities of key fission gases (I, Te, Xe and Kr) in pyrocarbon (PyC) coatings. Sufficient data are needed to develop and refine effective diffusivity correlations with uncertainties [<10x] at 95% confidence level.

It is expected that Pu isotopes and the Minor Actinides will be completely retained by intact SiC coatings under normal operation and core conduction cooldown conditions to at least 1600 °C; however, this expectation should be confirmed experimentally, especially for peak burnup and fast fluence.

1.4 Parameters/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications as appropriate:

Parameter	Value
Test Article	TRISO-coated DF and TF particles
DF burnup range	[0 to 85]% FIMA
TF burnup range	[0 to 85]% FIMA
DF Peak fast fluence	[8 x 10^{25} n/m^2 (E >29 fJ)]
TF Peak fast fluence	[8 x 10^{25} n/m^2 (E >29 fJ)]
System pressure[*]	

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[*] A lower pressure may be used to satisfy DDN if adequate technical justification is provided.

Normal condition	7.1 MPa (70 atm)
Off-normal conditions	0.1 to 7.1 MPa (1 to 70 atm)
Fuel temperatures/time	
Normal condition	[800 to 1400 C]/≤36 months
Off-normal conditions (core conduction cooldown)	
Pressurized (wet)	[≤1400 ^{o}C]/≤100 h
Depressurized	[1200 to 2000 ^{o}C] /100 to 1000h
Environment	
Normal conditions	He/Ne(in-pile tests)
Off-normal conditions	He; He/H_2O/CO/H_2; He/CO/N_2
Coolant/Impurity Levels	
Normal condition (Design):	Total oxidants [≤7Pa (1ppmv)]
H_2O	[0.07 to 0.7 Pa (0.01 to 0.1 ppmv)]
CO	[1.5 to 6 Pa (0.2 to 8 ppmv)]
CO_2	[0.2 to 1.5 Pa 0.03 to 0.2 ppmv)]
H_2	[3 to 10 Pa (0.5 to 1.5 ppmv)]
CH_4	[0.3 to1.5 Pa (0.5 to 1.5 ppmv)]
N_2	[3 Pa (0.4 ppmv)]
Pressurized condition	
H_2O	[0 to TBD KPa (0 to TBD atm)]
Depressurized condition	
CO	[0 to 35 KPa (0 to 0.35 atm)]
N_2	[0 to 65 KPa (0 to 0.65 atm)]
Fission gases of interest	I > Kr > Te > Xe
Fission metals of interest	Ag >> Cs > Sr

2. DESIGNER'S ALTERNATIVES

The following alternatives have been considered:

2.1 Use FRG correlations for radionuclide diffusivities in coatings which were derived from data taken on low-burnup German particles and assume the data are applicable to US-made, high-burnup DF/TF fuel.

2.2 Assume no retention of Ag and complete retention of Cs and Sr by SiC coatings, and assume no gas retention by the OPyC on particles with failed or defective SiC coatings.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Determine the effective diffusivities of Cs, Ag, and Sr in SiC coatings and of Kr, Xe, I, and Te in OPyC coatings of irradiated, US-made, DF/TF particles. Correlate these diffusivities as a function of temperature, fluence, and as-manufactured coating attributes. Given the high yields of silver isotopes from Pu fissions in DF/TF fuel, the assumption of complete Ag release from the core is judged to be unacceptable for a direct-cycle DB-MHR.

The available data suggest that the diffusivities of volatile fission products in SiC and OPyC can be strongly dependent upon the physical structure of the coatings which can be influenced by the coating process parameters and particle irradiation history. Consequently, the most reliable data would be obtained from TRISO particles manufactured to DB-MHR product specifications and irradiated under conditions representative of DB-MHR cores.

4. SCHEDULE REQUIREMENTS

Preliminary data [6] months before start of Preliminary Design and final data [12] months prior to completion of Final Design.

5. PRIORITY

Urgency: TBD
Cost benefit: H
Uncertainty in existing data: M
Importance of new data: H (for silver)

6. CONSEQUENCES OF NONEXECUTION AND FALLBACK POSITION

Alternative 2.1 along with a more conservative fuel and core design to account for the uncertainties resulting from deriving the retention characteristics of high-burnup DF/TF particles from low-burnup, German LEU UO_2 data. Failure to fully exploit the inherent retentivity of TRISO particles will necessitate a more conservative fuel and core design which could include unnecessarily restrictive limits on fuel temperatures during normal operation and core conduction cooldown transients.

7. REFERENCES

1. "Fuel Performance and Fission Product Behavior in Gas Cooled Reactors," IAEA-TECDOC-978, International Atomic Energy Agency, November 1997, pp. 437-447.

2. Minato, K., et al., "Fission Product Release Behavior of Individual Coated Fuel Particles for High-Temperature Gas-Cooled Reactors," Nucl. Tech., **131**, 36-47 (July 2000).

3. Foerthmann, R., and E. Vekens, "Transportverhalten von Plutonium and Americium in niedrig angereicherten Beschichteten Brennstoffteilchen bei hohen Bestrahlungstemperaturen," Juel-1811, KFA, Oktober, 1982.

4. e. Silva, A.T., "Experimentalle Untersuchungen der Aktinidenfreisetzung aus Brennstoffteilchen fuer Hochtemperaturreaktoren," Juel-1833, Maerz 1983.

5. Baldwin, N.L., et al., "Diffusion of Plutonium in High-Temperature Pyrolytic Carbon," Nucl. Tech., **37**, 353-357 (1978).

Originator	Date
Engineering Manager	Date
Project Manager	Date

DATE: 09/27/02

"AAA" Fuel Program
FISSION PRODUCT DIFFUSIVITIES/SORPTIVITIES IN GRAPHITE
DDN RTW.03.05

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The fuel element graphite can significantly attenuate the release of fission metals and should preclude the release of actinides from the core during normal operation and during core conduction cooldown transients; consequently, the transport properties of fission metals in graphite must be determined as a function of environmental and irradiation conditions (actinide retention is addressed in DDN RTW.03.06).

1.1 Summary of Function/Assumptions

"Retain Radionuclides in Core Graphite," Assumption: Transport of radionuclides in core graphite is adequately described by reference correlations to within a factor of [<10x] at 95% confidence.

"Protect the Capability to Retain Radionuclides in Core Graphite," Assumption: Transport of fission metals in core graphite during core conduction cooldown transients is adequately described by reference correlations to within a factor of [TBD] at 95% confidence.

"Retain Radionuclides in Core Graphite," Assumption: Transport of fission metals in core graphite under accident conditions is adequately described by reference models to within [<10x] at 95% confidence.

1.2 Current Data Base Summary

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 (e.g., Ref. 1). The correlations for Cs, Sr, and Pu sorptivities on graphite are derived largely from measurements on unirradiated graphites, but there are limited data for Cs and Sr on irradiated graphite and irradiated fuel compact matrix material. The available data indicate that the transport of Cs, Sr, and Ag in graphite is strongly affected by neutron irradiation. The current correlation for Ag diffusivity in irradiated H-451 was inferred from the measured Ag diffusivity in German A3 matrix.

The sorptivities of Cs and Sr on H-451 and H-327 graphites and over petroleum pitch matrix materials have been measured in the laboratory at partial pressures

>10^{-10} atm. The sorptivities of Cs and Sr 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; the sorptivity of pitch matrix is independent of fast fluence. The sorptivities Cs, Sr and Ag on German thermosetting-resin matrix, including A3 matrix, have been measured and may apply to candidate U.S. resin matrix materials (e.g., Ref. 2). There are limited laboratory data that indicate the vapor pressure of Cs over graphite increases in the presence of coolant impurities and as a consequence of partial graphite oxidation. Dragon Project data imply that Ag transport through graphite may be reduced dramatically by elevated system pressures.

1.3 Data Needed

Correlations/models for the diffusivities and sorptivities of Ag, Cs, and Sr in fuel-compact matrix and core 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 dry and wet core conduction cooldown conditions. Sufficient data are needed to develop and refine diffusivity and sorptivity correlations with uncertainties <10x at 95% confidence.

1.4 Parameters/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications as appropriate:

Parameter	Value
Test Article	[Thermosetting resin matrix; “H-451” graphite]
DF Peak fast fluence	[8 x 10^{25} n/m^2 (E >29 fJ)]
TF Peak fast fluence	[8 x 10^{25} n/m^2 (E >29 fJ)]
System pressure[*]	
Normal condition	7.1 MPa (70 atm)
Off-normal conditions	0.1 to 7.1 MPa (1 to 70 atm)
Graphite temperatures/time	
Normal condition	[500 to 1300 °C]/≤36 months
Off-normal conditions (core conduction cooldown)	
Pressurized (wet)	[≤1400 °C]/≤100 h
Depressurized	[1200 to 2000 °C] /100 to 1000h
Environment	
Normal conditions	He/Ne(in-pile tests)
Off-normal conditions	He; $He/H_2O/CO/H_2$; $He/CO/N_2$
Coolant/Impurity Levels	
Normal condition (Design):	Total oxidants [≤7Pa (1ppmv)]
H_2O	[0.07 to 0.7 Pa (0.01 to 0.1 ppmv)]
CO	[1.5 to 6 Pa (0.2 to 8 ppmv)]

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[*] A lower pressure may be used to satisfy DDN if adequate technical justification is provided.

CO_2	[0.2 to 1.5 Pa 0.03 to 0.2 ppmv)]
H_2	[3 to 10 Pa (0.5 to 1.5 ppmv)]
CH_4	[0.3 to1.5 Pa (0.5 to 1.5 ppmv)]
N_2	[3 Pa (0.4 ppmv)]
Pressurized condition	
H_2O	[1 to TBD KPa (0.01 to TBD atm)]
Depressurized condition	
CO	[0 to 35 KPa (0 to 0.35 atm)]
N_2	[0 to 65 KPa (0 to 0.65 atm)]
Range of graphite burnoff	0.1% to 10%
Radionuclides of interest	Ag, Cs > Sr

2. DESIGNER'S ALTERNATIVES

The following alternatives have been considered:

2.1 Use the current reference correlations which have very large uncertainties.

2.2 Use the reference German correlations for transport in pebble matrix and FRG graphites which do not explicitly treat irradiation or environmental effects.

2.3 Do not take credit for the core graphite as a barrier to fission metal release and rely exclusively on the SiC coating for control.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Complete the measurement and modeling of fission metal transport in fuel-compact matrix and core graphite, and establish a correlation that explicitly accounts for the effects of temperature, fluence, coolant impurities, graphite burnoff and, if appropriate, system pressure and fission metal concentration.

The core graphite should be a very significant barrier to the release of fission metals; however, the reference correlations have very large uncertainties because many of the apparent variables cited above are not treated explicitly and because the correlations are based largely on measurements made on unirradiated nonreference graphites (hence, the rejection of Alternative 2.2).

Alternative 2.3 is rejected because the core graphite is predicted to attenuate the releases of fission metals by orders of magnitude during core conduction cooldown accidents.

4. SCHEDULE REQUIREMENTS

Preliminary data [6] months before start of Preliminary Design and final data [12] months prior to completion of Final Design.

5. PRIORITY

Urgency: TBD
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H

6. CONSEQUENCES OF NONEXECUTION AND FALLBACK POSITION

Choose alternative 2.1 and accept the very large uncertainties in the reference correlations which result, at least in part, from not explicitly considering irradiation or environmental effects. The risk is that safety reviewers may not give credit for the substantial attenuation of fission metal release by the core graphite during normal operation and core conduction cooldown accidents. If no credit is taken for the attenuation of fission metal release by the core graphite, the retention requirements imposed upon the fuel particle coatings become correspondingly more stringent.

7. REFERENCES

1. "Fuel Performance and Fission Product Behavior in Gas Cooled Reactors," IAEA-TECDOC-978, International Atomic Energy Agency, November 1997, pp. 447-471.

2. Moormann, R., and K. Verfondern, "Methodik umfassender probabilistischer Shicherheitsanalysen fuer zukuenftige HTR-Anlagenkonzepte – Ein Statusbericht (Stand 1986), Band 3: Spaltproduktfreisetzung," KFA Report Juel-Spez-388, Vol. 3, Research Center Juelich, 1987.

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

“AAA” Fuel Program
MINOR ACTINIDE DIFFUSIVITIES/SORPTIVITIES IN GRAPHITE
DDN RTW.03.06

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The fuel element graphite should preclude the release of actinides from the core during normal operation and during core conduction cooldown transients; consequently, the transport properties of actinides in graphite must be determined as a function of environmental and irradiation conditions (fission product retention by graphite was addressed in DDN RTW.03.05).

1.1 Summary of Function/Assumptions

"Retain Radionuclides in Core Graphite," Assumption: Transport of radionuclides in core graphite is adequately described by reference correlations to within a factor of [<10x] at 95% confidence.

"Control Transport in Primary Circuit," Assumption: The available data, design methods and computer codes for predicting transport in primary circuit are accurate to within [10x] at 95% confidence.

"Protect the Capability to Retain Radionuclides in Core Graphite," Assumption: Transport of fission metals in core graphite during core conduction cooldown transients is adequately described by reference correlations to within a factor of [TBD] at 95% confidence.

"Retain Radionuclides in Core Graphite," Assumption: Transport of fission metals in core graphite under accident conditions is adequately described by reference models to within [<10x] at 95% confidence.

1.2 Current Data Base Summary

The diffusivities of plutonium in unirradiated H-451 graphite has been measured up to 1350 °C (Ref. 1), and the desorption pressure of Pu sorbed on H-451 has been measured up to 1350 °C (Ref. 2). Based upon these measured transport properties, the release of Pu from the graphite into the primary coolant was predicted to be negligible for both normal operation and depressurized core conduction cooldown conditions. In addition, the sorption of PuC on H-451 graphite (Ref. 3) and uranium diffusion in H-451 graphite have been measured (Ref. 4).

The transport properties of actinides in compact-matrix material and of Am and Cm in graphite have not been measured.

1.3 Data Needed

Correlations/models are needed for the diffusivities and sorptivities of Pu and other actinides (Am, Cm and Np) in fuel-compact matrix and core graphites as a function of temperature, fast fluence, and, as appropriate, coolant impurities, and the extent of graphite oxidation under normal operating and dry and wet core conduction cooldown conditions. Sufficient data are needed to develop and refine diffusivity and sorptivity correlations with uncertainties [<10x] at 95% confidence.

1.4 Parameters/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications as appropriate:

Parameter	Value
Test Article	[Thermosetting resin matrix; “H-451” graphite]
DF Peak fast fluence	[8 x 10^{25} n/m^2 (E >29 fJ)]
TF Peak fast fluence	[8 x 10^{25} n/m^2 (E >29 fJ)]
System pressure*	
Normal condition	7.1 MPa (70 atm)
Off-normal conditions	0.1 to 7.1 MPa (1 to 70 atm)
Graphite temperatures/time	
Normal condition	[500 to 1300 °C]/≤36 months
Off-normal conditions (core conduction cooldown)	
Pressurized (wet)	[≤1400 °C]/≤100 h
Depressurized	[1200 to 2000 °C] /100 to 1000h
Environment	
Normal conditions	He/Ne(in-pile tests)
Off-normal conditions	He; He/H_2O/CO/H_2; He/CO/N_2
Coolant/Impurity Levels	
Normal condition (Design):	Total oxidants [≤7Pa (1ppmv)]
H_2O	[0.07 to 0.7 Pa (0.01 to 0.1 ppmv)]
CO	[1.5 to 6 Pa (0.2 to 8 ppmv)]
CO_2	[0.2 to 1.5 Pa 0.03 to 0.2 ppmv)]
H_2	[3 to 10 Pa (0.5 to 1.5 ppmv)]
CH_4	[0.3 to1.5 Pa (0.5 to 1.5 ppmv)]
N_2	[3 Pa (0.4 ppmv)]
Pressurized condition	
H_2O	[0 to TBD KPa (0 to TBD atm)]
Depressurized condition	[0 to 35 KPa (0 to 0.35 atm)]

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*A lower pressure may be used to satisfy DDN if adequate technical justification is provided.

CO	[0 to 65 KPa (0 to 0.65 atm)]
N_2	
Range of graphite burnoff	0.1% to 10%
Radionuclides of interest	Am, Pu > Cm, Np

2. DESIGNER'S ALTERNATIVES

The following alternatives have been considered:

2.1 Use the current Pu correlations for all minor actinides with the exception of Am; increase the desorption pressure of Am over graphite by the ratio of the elemental vapor pressures.

2.2 Do not take credit for the core graphite as a barrier to actinide release and rely exclusively on the SiC coating for control.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Complete the measurement and modeling of actinide transport in fuel-compact matrix and core graphites, and establish a correlation that explicitly accounts for the effects of temperature, fluence, coolant impurities, graphite burnoff and, if appropriate, system pressure and actinide concentration.

The core graphite should be a very significant barrier to the release of Pu and higher actinides however, the Pu correlations have large uncertainties because many of the apparent variables cited above are not treated explicitly and because the correlations are based largely on measurements made on unirradiated graphites; the transport properties of the higher actinides in matrix and graphite have not been measured.

Alternative 2.2 is rejected because the core graphite is expected to attenuate the releases of actinides by orders of magnitude during normal operation and core conduction cooldown accidents.

4. SCHEDULE REQUIREMENTS

Preliminary data [6] months before start of Preliminary Design and final data [12] months prior to completion of Final Design.

5. PRIORITY

Urgency: TBD
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H

6. CONSEQUENCES OF NONEXECUTION AND FALLBACK POSITION

Alternative 2.1 offers a reasonable basis for constructing actinide retention models in graphite. Retention in compact matrix would be approximated by known differences in surface area per unit volume (i.e., no credit would be taken for the expected, larger intrinsic sorptivity of the matrix). Regulators may require additional conservatism and in the limit may not give credit for actinide retention by the graphite (Alternative 2.2).

Choice of Alternative 2.2 would place an additional heavy burden on the SiC as the sole barrier to actinide release from the core during normal operation and accidents. The resulting, allowable as-manufactured contamination and inservice failure limits would be sufficiently low to constitute a serious feasibility issue.

7. REFERENCES

1. Godsey, T.T., et al., "Diffusion of Plutonium in H-451 Graphite at 1000 to 1350 °C," HTGR-85-033, ORNL, January 1986.

2. Fellows, R.L., et al., "Plutonium Transport in High-Temperature Reactor Graphite," HTGR-85-162, ORNL, February 1987.

3. Tallent, O.K., et al., "Vapor Pressure of Plutonium Carbide Adsorbed on Graphite", ORNL/TM-9161, September 1984.

4. Tallent O.K., et al., "Uranium Diffusion in H-451 Graphite", Nucl. Tech., **68**, 336-343, 1985.

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

"AAA" Fuel Program
FISSION GAS RELEASE VALIDATION DATA
DDN RTW.03.07

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The design methods and codes used to predict fission gas release from the core (SURVEY for normal operation and SORS and OXIDE for accidents), including the radiologically important radioiodines, must be validated to have the specified predictive accuracies for normal operating conditions, for core conduction cooldown transients, and [for H_2O ingress transients] representative of the DB-MHR. Integral test data, representative of the DB-MHR, are required for comparison with predictions of fission gas release, including iodine release, during normal operation and postulated accident conditions. The data must be independent of that originally used to develop the predictive methods.

1.1 Summary of Function/Assumptions

"Control Transport from Core," Assumption: The existing design methods and computer codes for calculating fuel failure and fission gas release from prismatic cores are accurate to within [4x] at 95% confidence.

"Protect the Capability to Control Transport from Core," Assumption: The existing design methods and computer codes for calculating gas release, including iodine release, from a prismatic core during transients are accurate to within [TBD] at 95% confidence

"Control Transport from Core," Assumption: Validated methods and data are available to adequately assess fuel failure, fission product release (from the core), and release from the primary circuit under accident conditions [to within 4x at 95% confidence].

1.2 Current Data Base Summary

The validity of the reference design methods for predicting fission gas release during normal operation has been assessed by applying them to FSV, Peach Bottom, and several irradiation capsules. The noble gas release from FSV at end-of-life was overpredicted by about a factor of two; the cause of the overprediction is ambiguous: fuel failure may have been overpredicted, or the long-term, in-pile effect of hydrolysis may be less severe than observed in lab tests, or a combination of both these effects (Ref. 1). The noble gas release from Peach Bottom Core 2 at end-of-life was underpredicted by a factor of two or three; however, the dominant source of gas release was heavy-metal contamination so not all the features of the gas release methodology were tested (Ref. 2). Both FSV and Peach Bottom Core 2 contained carbide fuel.

The fission gas release from irradiation capsules containing LEU UCO/ThO_2 fuel is generally predicted to within a factor of about five. However, these capsules operated dry so the hydrolysis model was not tested. Moreover, there is inherent ambiguity in these data since the fuel failure fraction is not known with high accuracy independent of the gas release data.

The fission gas release from TRISO-P in the three NPR capsules and the HRB-21 capsule was originally grossly under predicted with the reference design methods for the following reasons: (1) the FDDM failure models, which were based upon conventional 5-layer TRISO fuel, did not properly account for the coating failure mechanisms introduced for with TRISO-P fuel, and (2) the FDDM fission gas release model did not account for the large burnup dependence observed for release from failed HEU UCO particles at burnups >~30% FIMA (Ref. 3).

Considerable data on fission gas release from LEU UCO fuel are available from the COMEDIE BD-1 test. The Kr-85m R/B at end-of-life was predicted to within 2x, but the Xe-133 R/B was underpredicted by 5x, compared to the accuracy goal of 4x; in general, the dependence of R/B on isotope half life was greater than predicted (Ref. 4).

The validity of the transient gas release model used to analyze core conduction cooldown transients has not been rigorously assessed.

The validity of the reference design methods for predicting fission gas release from Pu cores has not been assessed. There are no available data which are directly suitable for the validation of the design methods that will be developed to predict fission gas release from DB-MHR cores containing DF and TF fuel.

1.3 Data Needed

An experimental data base is needed to assess the validity of the integrated models and core-survey codes used to predict fission gas release (Kr, Xe, I and Te) from the core with reference DF/TF fuel during normal operation (SURVEY) and under transient conditions (SORS) in order to assure that the predictive methods are accurate to within [4x] at 95% confidence. Particular attention must be given to effects of hydrolysis during steady-state power operation and to the transient release of iodines and noble gases under dry and wet core conduction cooldown conditions.

The data for assessing the overall accuracy of the gas release methodology must be independent of the data from which the individual correlations in the overall design method were originally derived (fuel failure models, gas release models for contamination and failed particles, etc.). [Quality assurance for the test programs must be in accordance with the requirements for validation testing as defined by NQA-1-1994.]

1.4 Parameters/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications as appropriate:

Normal Operating Conditions

Parameter	Value
Reference fuel	TRISO-coated DF/TF in compacts
Fuel burnup range	[0 to 85]% FIMA
Peak fast fluence	8 x 10^{25} n/m^2 (E >29 fJ)
System pressure[*]	7.1 MPa (70 atm)
Fuel temperatures/time	[800 to 1400 C]/≤36 months
He velocity (coolant channel)	>10 m/s
Environment	He/Ne(in-pile tests)
Coolant/Impurity Levels	
Normal operation (Design):	Total oxidants [≤7Pa (1ppmv)]
H_2O	[0.07 to 0.7 Pa (0.01 to 0.1 ppmv)]
CO	[1.5 to 6 Pa (0.2 to 8 ppmv)]
CO_2	[0.2 to 1.5 Pa 0.03 to 0.2 ppmv)]
H_2	[3 to 10 Pa (0.5 to 1.5 ppmv)]
CH_4	[0.3 to1.5 Pa (0.5 to 1.5 ppmv)]
N_2	[3 Pa (0.4 ppmv)]
Expected failure fraction	[10^{-5} to 10^{-3}]

Off-Normal Conditions

Reference transients	Pressurized conduction cooldown with and without water ingress Depressurized conduction
Environment	He; He/CO/H_2; He/CO/N_2
Fuel temperatures/time	
Pressurized	[1300 °C]/up to 100 h
Depressurized	[1200 to 1600 °C] from 300 to 1000 h [1600 to 2000 °C] from 100 to 200 h
System pressure	0.1 to 7.1 MPa (1 to 70 atm)
Coolant Impurity Levels	
Pressurized conditions	[0 to TBD MPa (0 to TBD atm) H_2O]
Depressurized	[0 to 35 KPa (0 to 0.35 atm) CO] [0 to 65 KPa (0 to 0.65 atm) N_2]
Fission gases of interest	I , Te > Kr, Xe

2. DESIGNER'S ALTERNATIVES

The following alternatives have been considered:

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[*] A lower pressure may be used to satisfy DDN if adequate technical justification is provided.

2.1 Rely upon existing, international in-pile data for uranium and thorium fuels to provide validation of fission gas release methods.

2.2 Rely upon comparisons of design codes with analytical solutions and other validated transport codes, including the FRG codes as available, through a series of benchmark calculations.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Obtain fission gas release data for DT/TF from in-pile experiments and postirradiation heating tests, and compare predicted and observed results and assess accuracy of design methods. Stringent limits on fission product release from the core have been specified for the DB-MHR. With these very tight limits on core release, large design margins to compensate for the current uncertainties in the fission gas release methods can not be tolerated; therefore, validation of the gas release methods is essential to avoid major retrofitting in the final design phase.

The alternatives are rejected because licensing requires validation of design methods using reference fuel in realistic integral tests. Use of nonprototypic fuel and conditions can apply only if there is a solid theoretical connection between the nonprototype and prototype; this linkage is dubious in the present case because of the uniqueness of the DF and TF fuel.

In particular, Alternative 2.1 is rejected because there are no definitive data for gas release for high burnup PuO_x fuels, especially under core conduction cooldown conditions; validation data for TRISO-coated U and Th fuel forms are supportive, but some explicit DF/TF data are essential. Alternative 2.2 is rejected because there are no formally validated codes for fission gas release from coated-particle fuel to serve as a benchmark. In addition, neither alternative meets in the intent of NQA-1-1994 which is assumed to apply to the licensing of the DB-MHR.

4. SCHEDULE REQUIREMENTS

Final data 12 months prior to completion of Final Design.

5. PRIORITY

Urgency: TBD
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H

6. CONSEQUENCES OF NONEXECUTION AND FALLBACK POSITION

The first alternative with the necessity of added conservatism in the design to compensate for calculational uncertainties, resulting from reliance upon integral test data for nonreference fuels (>>10x). A weakened safety review position as a consequence of not being able to answer the obvious question of how well do the design methods predict the fission gas release from reference DF and TF particles. The ultimate consequence could be the necessity of a major retrofitting in the final design phase to meet stringent technical specifications on circulating activity and/or the imposition of a high-pressure containment and active cooling systems on the design.

7. REFERENCES

1. Baxter, A.M., et al., "FSV Experience in Support of the GT-MHR Reactor Physics, Fuel Performance, and Graphite," GA-A21925, General Atomics, November 1994.

2. Steward, K. P., "Final Summary Report on the Peach Bottom End-of-Life Program," DOE Report GA-Al4404, General Atomic Company, 1978.

3. Richards, M.B., "Status Report: Empirical Fuel Performance Model based upon PIE Data from Capsules HRB-21, NPR-1, and NPR-1A," CEGA-002952, General Atomics, October 1993.

4. Medwid, W., and A. Gillespie, "COMEDIE BD-1 Test Evaluation Report," DOE-HTGR-88552, Rev. 0, General Atomics, 1993.

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

"AAA" Fuel Program
FISSION METAL AND ACTINIDE RELEASE VALIDATION DATA
DDN RTW.03.08

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The design methods and codes used to predict the release of fission metals and actinides from the core (TRAFIC/COPAR and TRAMP/COPAR for normal operation and SORS for accidents) must be validated to have the specified predictive accuracies for normal operating conditions and for core conduction cooldown transients. Integral test data, representative of the DB-MHR core, are required for comparison with predictions of fission metal release, including Pu and higher actinide release, during normal operation and postulated accident conditions. The data must be independent of that originally used to develop the predictive methods.

1.1 Summary of Function/Assumptions

"Control Transport from Core," Assumption: The existing design methods and computer codes for calculating fission metal release from a prismatic core are accurate to within [10x] at 95% confidence.

"Protect the Capability to Control Transport from Core," Assumption: The existing design methods and computer codes for calculating fission metal release from a prismatic core during core conduction cooldown transients, including the effects of redeposition in the colder portions of the core, are accurate to within a factor of [TBD] at 95% confidence.

"Control Transport from Core," Assumption: Validated methods and data are available to adequately assess fuel failure, fission product transport, and release from the primary circuit under accident conditions [to within 10x at 95% confidence].

1.2 Current Data Base Summary

The validity of the methods for predicting fission metal release during normal operation have been assessed by applying them to predict the observed metal release in Peach Bottom Core 2 (Ref. 1) and FSV (Ref. 2) and in irradiation capsules and in-pile loops (e.g., SSL1, SSL2, Idylle 03, the four CPL2 loops, and R2 K13). Most of the available data are for the Cs isotopes with a small amount of Ag and Sr 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. The cause of the underpredictions is ambiguous because the SiC defect fractions and the particle failure fractions are typically not well known; however, there is strong circumstantial evidence suggesting that the transport across the fuel compact/fuel block gap and the transport in the graphite web are not properly modeled.

Data on fission metal release from LEU UCO fuel are available from the COMEDIE BD-1 test. (The value of the BD-1 metal release data was seriously compromised by the failure to perform the planned PIE of the fuel element and reflector element.) The measured Cs-137/Cs-134 plateout inventories were underpredicted by nearly 30x using the FDDM UCO kernel release correlation; Cs release was predicted within 2x using the German UO_2 correlation, well within the specified accuracy goal of 10x (Ref. 3).

The validity of the methods for predicting fission metal release during core conduction cooldown transients have not been assessed systematically.

The validity of the reference design methods for predicting fission metal release and actinide release from Pu cores has not been assessed. There are no available data which are directly suitable for the validation of the design methods that will be developed to predict fission metal release and actinide release from DB-MHR cores containing DF and TF fuel.

1.3 Data Needed

An experimental data base is needed to validate the integrated models and core-survey codes used to predict fission metal release and actinide release from the DB-MHR core with DF/TF during normal operation (TRAFIC and COPAR) and under dry and wet core conduction cooldown conditions (SORS) in order to assure that the predictive methods are accurate to within [10x] at 95% confidence. Particular attention must be given to the effects of irradiation and environment on the transport of fission metals in core graphite.

Certain test data must investigate the prediction that fission metals released from the hotter portions of the core redeposit in the colder portions of the core, especially on the upper reflector; this redeposition is projected to attenuate the release of Ag, Cs and Sr from the core by three orders-of-magnitude or more during core conduction cooldown events. Moreover, data are needed to confirm the prediction that Pu and higher actinides outside of intact TRISO particles are completely retained by the core graphite.

The data for assessing the overall accuracy of the metal release methodology must be independent of the data from which the individual correlations in the overall method were originally derived (fuel failure models, graphite diffusivities and sorptivities, etc.). [Quality assurance for the test programs must be in accordance with the requirements for validation testing as defined by NQA-1-1994.]

1.4 Parameters/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications as appropriate:

Normal Operating Conditions

Parameter	Value
Reference fuel	TRISO-coated DF/TF in compacts
Fuel burnup range	[0 to 85]% FIMA
Peak fast fluence	8 x 10^{25} n/m^2 (E >29 fJ)
System pressure[*]	0.1 to 7.1 MPa (1 to 70 atm)
Fuel temperatures/time	[800 to 1400 C]/≤36 months
He velocity (coolant channel)	>10 m/s
Environment	He/Ne(in-pile tests)
Coolant/Impurity Levels	
Normal operation (Design):	Total oxidants [≤7Pa (1ppmv)]
H_2O	[0.07 to 0.7 Pa (0.01 to 0.1 ppmv)]
CO	[1.5 to 6 Pa (0.2 to 8 ppmv)]
CO_2	[0.2 to 1.5 Pa 0.03 to 0.2 ppmv)]
H_2	[3 to 10 Pa (0.5 to 1.5 ppmv)]
CH_4	[0.3 to1.5 Pa (0.5 to 1.5 ppmv)]
N_2	[3 Pa (0.4 ppmv)]
Expected failure fraction	[10^{-5} to 10^{-3}]

Off-Normal Conditions

Reference transients	Pressurized conduction cooldown with and without water ingress Depressurized conduction
Environment	He; He/CO/H_2; He/CO/N_2
Fuel temperatures/time	
Pressurized	[1300 °C]/up to 100 h
Depressurized	[1200 to 1600 °C] from 300 to 1000 h [1600 to 2000 °C] from 100 to 200 h
System pressure	0.1 to 7.1 MPa (1 to 70 atm)
Coolant Impurity Levels	
Pressurized conditions	[0 to TBD MPa (0 to TBD atm) H_2O]
Depressurized	[0 to 35 KPa (0 to 0.35 atm)] CO [0 to 65 KPa (0 to 0.65 atm)] N_2

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[*] A lower pressure may be used to satisfy DDN if adequate technical justification is provided.

Radionuclides of interest Ag, Cs > Am, Pu > Sr, Cm, Np

2. DESIGNER'S ALTERNATIVES

The following alternatives have been considered:

2.1 Rely upon existing in-pile data for uranium and thorium fuels to provide validation of fission metal release and actinide release methods.

2.2 Rely upon comparisons of design codes with analytical solutions and other validated transport codes, including the FRG codes as available, through a series of benchmark calculations.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Obtain fission metal release and actinide release data for DT/TF from in-pile experiments and postirradiation heating tests. Compare predicted and observed results and assess accuracy of design methods. Stringent limits on radionuclide release from the core have been specified for the DB-MHR. With these very tight limits on core release, large design margins to compensate for the current uncertainties in the release methods can not be tolerated. Therefore, validation of the metal release methods, including Pu and higher actinide release, is essential to avoid major retrofitting in the final design phase.

The alternatives are rejected because licensing requires validation of design methods using reference fuel in realistic integral tests. Use of nonprototypic fuel and conditions can apply only if there is a solid theoretical connection between the nonprototype and prototype; this linkage is dubious in the present case because of the uniqueness of the DF and TF fuel.

In particular, Alternative 2.1 is rejected because there are no definitive data for metal release, including Pu and higher actinide release, for high burnup PuO_x fuels, especially under core conduction cooldown conditions; validation data for TRISO-coated U and Th fuel forms are supportive, but some explicit DF/TF data are essential. Alternative 2.2 is rejected because there are no formally validated codes for metal release from coated-particle fuel to serve as a benchmark. In addition, neither alternative meets in the intent of NQA-1-1994 which is assumed to apply to the licensing of the DB-MHR.

4. SCHEDULE REQUIREMENTS

Final data 12 months prior to completion of Final Design.

5. PRIORITY

Urgency: TBD
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H

6. CONSEQUENCES OF NONEXECUTION AND FALLBACK POSITION

The first alternative with the necessity of added conservatism in the design to compensate for calculational uncertainties (>>10x). A weakened safety review position as a consequence of not being able to answer the obvious question of how well do the design methods predict the radionuclide transport behavior from the DB-MHR core with DF and TF. The ultimate consequence could be the necessity of a major retrofitting in the final design phase to meet stringent technical specifications on circulating activity and/or the imposition of a high-pressure containment and active cooling systems on the design.

7. REFERENCES

1. Steward, K. P., "Final Summary Report on the Peach Bottom End-of-Life Program," DOE Report GA-Al4404, General Atomic Company, 1978.

2. Baxter, A.M., et al., "FSV Experience in Support of the GT-MHR Reactor Physics, Fuel Performance, and Graphite," GA-A21925, General Atomics, November 1994.

3. Medwid, W., and A. Gillespie, "COMEDIE BD-1 Test Evaluation Report," DOE-HTGR-88552, Rev. 0, General Atomics, 1993.

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Originator Date

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Engineering Manager Date

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Project Manager Date

DATE: 09/27/02

“AAA” Fuel Program
RADIONUCLIDE LEACHING RATES FROM DF KERNELS
DDN RTW.04.01

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The DB-MHR core design produces high-burnup, TRISO-coated DF particles. A fraction of these particles is expected to experience in-service and in-storage failure of all coatings, resulting in exposed kernels. In the event of a failure of the waste canister, designed to contain the DB-MHR spent fuel elements, a spent fuel element could be in contact with groundwater. In such a case, the radionuclide release rates to the groundwater will depend on the leaching rates from the fuel structures. For PC-MHR spent fuel, it was assumed that the radionuclide inventory in the exposed fuel kernels would be leached in a relatively short time compared with the disposal time scale. This estimate is based upon the German data for leaching rates of low-burnup UO_2 kernel radionuclide inventories in brine. For the DB-MHR high-burnup DF fuel, relevant data are needed for leaching rates of kernel radionuclide inventories in groundwater.

1.1 Summary of Functions/Assumptions

[TBD]

1.2 Current Data Base Summary

There are no explicit data for leaching rates for the exposed DF kernels nor for any other U.S. HTGR exposed fuel kernels. Several leaching tests have been run on LWR fuel (e.g., Refs 1 and 2), but these materials are not representative of the DB-MHR fuel particles. Germans have studied leaching of their HTR fuel, which is TRISO coated, like the DB-MHR fuel, but which has a different kernel composition and much lower burnup. The current data base consists primarily of data from the German leaching tests for the exposed irradiated HTR-AVR fuel kernels.

The German HTR-AVR leaching data are for UO_2 exposed kernels with the maximum burnup of 11% FIMA. Data from one such AVR test indicate that exposed HTR fuel kernels lose virtually the entire inventory of Cs-137 after approximately a year of leaching in brine (Ref. 3). There was no evidence of corrosion by the brine of the coatings on intact particles (The Germans typically leached samples in brine because their proposed repository was to be located in a salt dome formation.)

In another AVR test, HTR kernels were leached in brine at 90 °C (Ref. 4). Data from this test with UO_2 kernels indicated that 18% of Sb, 10% of Cs and 4% of U was leached out in 100 days. Higher leaching rates for Cs and Sb than for U are due their higher solubilities and, perhaps, to the probability that they accumulate at grain boundaries in the kernel which are more accessible to the leachant. The other actinides and lanthanides leached at rates lower than U. These include Pu, Am, Ru, Cm, Ce, Np, and Eu which leached at 1% to 3% in 100 days. These low leach rates may be due to their low solubilities. This leaching experiment was continued at 20 °C for 450 days. The results indicated that 3.5% of Cs and 1.2% U leached out after 450 days. Other elements, including Am, Ru, Ce, Cm, Pu, Sb, Np and Eu, leached out at 0.2% to 3%.

1.3 Data Needed

Correlations are needed describing radionuclide leaching rates from exposed DF kernels during the normal and off-normal (groundwater ingress) conditions. Data are needed from accelerated tests with failed, high burnup DF particles (i.e., particles with exposed kernels), including high-burnup Pu particles recovered from fuel test element FTE-13 irradiated in the Peach Bottom-1 reactor, which is currently in long-term storage. Correlations will cover the range of fuel temperatures for normal and off-normal conditions and a range of groundwater chemistries. The tests will necessarily be accelerated, but they should be of sufficient duration and at sufficiently elevated temperatures so that statistically significant quantities of the key radionuclides, including Pu isotopes and higher actinides, are dissolved and detected in the water.

1.4 Parameter/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications as appropriate:

Test Articles

Fuel	TRISO-coated DF particles and compacts
Matrix type	[Thermosetting resin]
Neutron poisons	PyC-coated B_4C PyC-coated Er_2O_3
Graphite body	Reference fuel-element graphite;[*] unit cell of prismatic element

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[*] It is assumed that the reference fuel-element graphite for the DB-MHR will be the functional equivalent of H-451.

Normal Repository Environment

Fuel burnup range	[0 to 85]% FIMA - Normal
Peak fast fluence	≤8 x 10^{25} n/m^2 (E > 29 fJ)
Repository pressure	1 atm
Residence time	≤10^6 years
Fuel element temperatures	[20° to 250°C]
Environment (Air)	[≤0.01 atm H_2O] [0.21 atm O_2] [0.79 atm N_2]
Expected exposed fuel fraction Normal conditions	[10^{-5} to 10^{-3}]

Off-Normal Conditions (Groundwater Ingress)

Groundwater chemistry	Yucca Mountain, [TBD}
Vapor-phase environment	O_2/N_2/H_2O/CO/H_2
Gas-phase impurity levels	[0 to 1.0] atm H_2O
Fuel-element temperatures	[20 - 100°C]
Groundwater contact time	TBD
Repository pressure	≥1 atm
Expected failure fraction Off-normal conditions	[10^{-4} to 10^{-2}]

2. DESIGNER'S ALTERNATIVES

Alternatives to the acquisition of the above data are:

2.1 Assume that the existing database for the low-burnup German UO_2 fuel can be used for high-burnup DB-MHR fuel, and add design margins to account for the uncertainties.

2.2 Assume complete and immediate dissolution of exposed DF/TF kernels when contacted by groundwater.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Measure the leaching rates for the DB-MHR DF/TF kernels in groundwater and in accelerated tests with failed particles, as a function of burnup, groundwater temperature, and groundwater chemistry. It is anticipated that some of these tests will be performed in high-pressure autoclaves to permit temperatures >>100 °C with liquid-phase water.

Alternative 2.1 cannot be justified technically because the differences in the fissile material (UO_2 for HTR vs. Pu/Np for DB-MHR DF), maximum burnup (11% vs. 85% FIMA), leachant composition (brine vs. groundwater), are too large to allow meaningful extrapolation of the HTR data. Alternative 2.2 is judged to be excessively conservative.

4. SCHEDULE REQUIREMENTS

TBD

5. PRIORITY

Urgency: TBD
Cost benefit: M
Uncertainty in existing data: H
Importance of new data: M

6. CONSEQUENCES OF NONEXECUTION AND FALLBACK POSITION

Use Alternative 2.2 and accept conservative estimates of radionuclide release from groundwater leaching of failed DF. The dominant licensing issue regarding source terms will likely be the extent of long-term, incremental coating failure (addressed in other DDNs). In the limit, complete coating failure and complete kernel dissolution would be assumed, and we would be forced to rely upon retardation and retention by engineered features within the repository (e.g., back-fill) and by the repository geosphere in order to show compliance with radionuclide release and dose regulations, which is the case for the disposal of whole, unprocessed LWR spent fuel.

7. REFERENCES

1. Gray, W. J., et al., "Parametric Study of LWR Spent Fuel Dissolution Kinetics," J. Nucl. Mater., **190**, p. 46, 1992.

2. Steward, S. A., et al., "Interlaboratory Comparison of UO_2 Dissolution Rates," Proceedings of the Fifth Annual International Conference on High Level Radioactive Waste Management, Vol. 4, p. 2597, 1994.

3. Kirch, N., U. Brinkmann, and H. Bruecher, "Storage and Disposal of HTR Fuel in FRG," Nucl. Eng. Des., **121**, 241 (1990).

4. Brodda, B. G. and E. R. Merz, "Leachability of Actinides and Fission Products From Spent HTR Fuel", Radiochimica Acta, **44/45**, 3 (1988).

Originator	Date
Engineering Manager	Date
Project Manager	Date

DATE: 09/27/02

"AAA" Fuel Program
RADIONUCLIDE LEACHING RATES FROM TF KERNELS
DDN RTW.04.02

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The DB-MHR core design produces high-burnup, TRISO-coated TF particles. A fraction of these particles is expected to experience in-service and in-storage failure of all coatings, resulting in exposed kernels. In the event of a failure of the waste canister, designed to contain the DB-MHR spent fuel elements, a spent fuel element could be in contact with groundwater. In such a case, the radionuclide release rates to the groundwater will depend on the leaching rates from the fuel structures. For PC-MHR spent fuel, it was assumed that the radionuclide inventory in the exposed fuel kernels would be leached in a relatively short time compared with the disposal time scale. This estimate is based upon the German data for leaching rates of low-burnup UO_2 kernel radionuclide inventories in brine. For the DB-MHR high-burnup TF fuel, relevant data are needed for leaching rates of kernel radionuclide inventories in groundwater.

1.1 Summary of Functions/Assumptions

[TBD]

1.2 Current Data Base Summary

There are no explicit data for leaching rates for the exposed TF kernels, or for any other U.S. HTGR exposed fuel kernels. Several leaching tests have been run on LWR fuel (e.g., Refs 1 and 2), but these materials are not representative of the DB-MHR fuel particles. The Germans have studied leaching of their HTR fuel, which is TRISO coated like DB-MHR fuel, but which has a different kernel composition and much lower burnup. The current data base consists primarily of data from the German leaching tests for the exposed irradiated HTR-AVR fuel kernels.

The German HTR-AVR leaching data are for UO_2 exposed kernels with the maximum burnup of 11% FIMA. Data from one such AVR test indicate that exposed HTR fuel kernels lost virtually the entire inventory of Cs-137 after approximately a year of leaching in brine (Ref. 3). There was no evidence of corrosion by the brine of the coatings on intact particles. (The Germans typically leached samples in brine because their proposed repository was to be located in a salt dome formation.)

In another AVR test, HTR kernels were leached in brine at 90 °C (Ref. 4). Data from this test with UO_2 kernels indicated that 18% of Sb, 10% of Cs and 4% of U was leached out in 100 days. Higher leaching rates for Cs and Sb than for U are due their higher solubilities and, perhaps, to the probability that they accumulate at grain boundaries in the kernel which are more accessible to the leachant. The other actinides and lanthanides leached at rates lower than U. These include Pu, Am, Ru, Cm, Ce, Np, and Eu which leached at 1% to 3% in 100 days. These low leach rates may be due to their low solubilities. This leaching experiment was continued at 20 °C for 450 days. The results indicated that 3.5% of Cs and 1.2% U leached out after 450 days. Other elements, including Am, Ru, Ce, Cm, Pu, Sb, Np and Eu, leached out at 0.2% to 3%.

1.3 Data Needed

Correlations are needed describing radionuclide leaching rates from exposed TF kernels during the normal and off-normal (groundwater ingress) conditions. Data are needed from accelerated tests with failed, high burnup TF particles (i.e., particles with exposed kernels), including high-burnup Pu particles recovered from fuel test element FTE-13 irradiated in the Peach Bottom-1 reactor, which is currently in long-term storage. Correlations will cover the range of fuel temperatures for normal and off-normal conditions and a range of groundwater chemistries. The tests will necessarily be accelerated, but they should be of sufficient duration and at sufficiently elevated temperatures so that statistically significant quantities of the key radionuclides, including Pu isotopes and higher actinides, are dissolved and detected in the water.

1.4 Parameter/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications as appropriate:

Test Articles

Fuel	TRISO-coated TF particles and compacts
Matrix type	[Thermosetting resin]
Neutron poisons	PyC-coated B_4C PyC-coated Er_2O_3
Graphite body	Reference fuel-element graphite;[*] unit cell of prismatic element

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[*] It is assumed that the reference fuel-element graphite for the DB-MHR will be the functional equivalent of H-451.

Normal Repository Environment

Fuel burnup range	[0 to 85]% FIMA - Normal
Peak fast fluence	$\leq 8 \times 10^{25}$ n/m^2 (E > 29 fJ)
Repository pressure	1 atm
Residence time	$\leq 10^6$ years
Fuel element temperatures	[20° to 250°C]
Environment	
(Air)	[$\leq$0.01 atm H_2O] [0.21 atm O_2] [0.79 atm N_2]
Expected exposed fuel fraction	
Normal conditions	[10^{-5} to 10^{-3}]

Off-Normal Conditions (Groundwater Ingress)

Groundwater chemistry	Yucca Mountain, [TBD}
Vapor-phase environment	$O_2/N_2/H_2O/CO/H_2$
Gas-phase impurity levels	[0 to 1.0] atm H_2O
Fuel-element temperatures	[20 - 100°C]
Groundwater contact time	TBD
Repository pressure	$\geq$1 atm
Expected failure fraction	
Off-normal conditions	[10^{-4} to 10^{-2}]

2. DESIGNER'S ALTERNATIVES

Alternatives to the acquisition of the above data are:

2.1 Assume that the existing database for low-burnup German UO_2 fuel can be used for high-burnup TF, and add design margins to account for the uncertainties.

2.2 Assume complete and immediate dissolution of exposed TF kernels when contacted by groundwater.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Measure the leaching rates for TF kernels in groundwater, in accelerated tests with failed particles, as a function of burnup, groundwater temperature, and groundwater chemistry. It is anticipated that some of these tests will be performed in high-pressure autoclaves to permit temperatures >>100 ^oC with liquid-phase water.

Alternative 2.1 cannot be justified technically because the differences in the fissile material (UO_2 for HTR vs. Pu/Am/Cm/Np for DB-MHR), maximum burnup (11% vs. 85% FIMA), leachant composition (brine vs. groundwater), are too large to allow meaningful extrapolation of the HTR data. Alternative 2.2 is judged to be excessively conservative.

4. SCHEDULE REQUIREMENTS

TBD

5. PRIORITY

Urgency: TBD
Cost benefit: M
Uncertainty in existing data: H
Importance of new data: M

6. CONSEQUENCES OF NONEXECUTION AND FALLBACK POSITION

Use Alternative 2.2 and accept conservative estimates of radionuclide release from groundwater leaching of failed TF. The dominant licensing issue regarding source terms will likely be the extent of long-term, incremental coating failure (addressed in other DDNs). In the limit, complete coating failure and complete kernel dissolution would be assumed, and we would be forced to rely upon retardation and retention by engineered features within the repository (e.g., back-fill) and by the repository geosphere in order to show compliance with radionuclide release and dose regulations, which is the case for the disposal of whole, unprocessed LWR spent fuel.

7. REFERENCES

1. Gray, W. J., et al., "Parametric Study of LWR Spent Fuel Dissolution Kinetics," J. Nucl. Mater., **190**, p. 46, 1992.

2. Steward, S. A., et al., "Interlaboratory Comparison of UO2 Dissolution Rates," Proceedings of the Fifth Annual International Conference on High Level Radioactive Waste Management, Vol. 4, p. 2597, 1994.

3. Kirch, N., U. Brinkmann, and H. Bruecher, "Storage and Disposal of HTR Fuel in FRG," Nucl. Eng. Des., **121**, 241 (1990).

4. Brodda, B. G. and E. R. Merz, "Leachability of Actinides and Fission Products From Spent HTR Fuel", Radiochimica Acta, **44/45**, 3 (1988).

Originator	Date
Engineering Manager	Date
Project Manager	Date

DATE: 09/27/02

"AAA" Fuel Program
LONG TERM MECHANICAL INTEGRITY OF STRESSED TRISO COATINGS
DDN RTW.04.03

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The TRISO coatings on spent DF/TF particles offer the potential for long-term (e.g., 10^5 years, or more) containment of radionuclides, well beyond that provided by the waste package. Confirmation of the long-term performance of TRISO coatings in a repository environment is needed to support repository performance assessments for comparison with regulatory criteria.

The long term, near-field containment of radionuclides by DB-MHR spent fuel depends in large part on the capability of the particle coatings to withstand for long times the stresses resulting from internal gas pressure (end-of-life fission gas and CO pressure plus helium pressure buildup due to alpha decay). This capability has been evaluated for PC-MHR spent fuel using static calculations of SiC hoop stress and quasi-static measurements of the Weibull distribution of SiC strength (Ref. 1). These methods and data have been used in the past to evaluate the ability of TRISO coatings to withstand internal pressure for short times relevant to fuel irradiation and temporary storage (e.g., up to ~30 years.) It is believed that these calculational methods and data are valid for determining the integrity of DB-MHR spent fuel coatings for the much longer times associated with permanent fuel disposal in a repository, but this must be confirmed.

1.1 Summary of Functions/Assumptions

[TBD]

1.2 Current Data Base Summary

There are extensive data which may be used to evaluate the short-term (e.g., up to ~30 years) capability of coated particles to withstand the level of internal gas pressure which may be expected to develop in DB-MHR spent fuel. These data include measurements of the crush strength of TRISO particles which can be used to evaluate SiC layer strength using Weibull failure theory (e.g., Ref. 2). There are also irradiation and heating data which directly demonstrate the capability of the coatings to withstand internal gas pressure as high as may be expected in DB-MHR spent fuel, for durations of up to several years. However, data which demonstrate that the TRISO coatings will withstand such pressure over time frames important to DB-MHR spent fuel disposal (e.g., 10^4 years or more) are not currently available.

1.3 Data Needed

Data are needed which would reduce the present uncertainty in the calculation of the capability of TRISO coatings to withstand the level of internal gas pressure expected to develop in spent DF/TF, over the time frames important to DB-MHR spent fuel disposal (e.g., 10^4 years or more). For example, the crush strength could be measured for coated particles that have been stored for different time periods, to determine the effect of storage time upon the crush strength. This may enable a correlation to be developed which would allow extrapolation of the crush strength to long time periods.

1.4 Parameters/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications as appropriate:

Test Articles

Fuel particles	TRISO-coated DF/TF in compacts
Matrix type	[Thermosetting resin]
Neutron poisons	PyC-coated B_4C PyC-coated Er_2O_3
Graphite body	Reference fuel-element graphite;[*] unit cell of prismatic element

Normal Repository Environment

Fuel burnup range	[0% to 86]% FIMA - Normal
Peak fast fluence	#8 x 10^{25} n/m^2 (E > 29 fJ)
Repository pressure	1 atm
Residence time	#10^6 years
Fuel element temperatures	[20E to 250EC]
Environment (Air)	[#0.01 atm H_2O] [0.21 atm O_2] [0.79 atm N_2]

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[*] It is assumed that the reference fuel-element graphite for the DB-MHR will be the functional equivalent of H-451.

Expected exposed fuel fraction	
Normal conditions	[10^{-5} to 10^{-3}]

Off-Normal Conditions (Groundwater Ingress)

Groundwater chemistry	Yucca Mountain, [TBD}
Vapor-Phase Environment	$O_2/N_2/H_2O/CO/H_2$
Vapor-phase Impurity Levels	[0 to 1.0] atm H_2O
Fuel-element temperatures	[20 - 100 ^oC]
Groundwater contact time	TBD
Repository pressure	$\geq$1 atm
Expected failure fraction	
Off-normal conditions	[10^{-4} to 10^{-2}]

2. DESIGNER'S ALTERNATIVES

Alternatives to the acquisition of the above data are:

2.1 Assume that the TRISO coatings do not retain fission products for the long times important to DB-MHR spent fuel disposal, and rely upon the retention characteristics of the waste package and the repository geosphere to meet applicable regulatory requirements (i.e., the LWR approach).

2.2 Assume that the existing data which have been used to evaluate the capability of TRISO coatings to withstand internal pressure for short times (relative to the times important for permanent fuel disposal in a repository) are sufficient for calculating the failure probabilities of the coatings for the much longer times associated with permanent fuel disposal.

2.3 Attempt to locate data on geological analogs to the SiC coatings (e.g., moissanite) which may provide insight into the long term strength and mechanical integrity of the coatings.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected approach is to perform tests to acquire the data specified in Section 1.2, while at the same time pursuing geologic analog data as specified in alternative 2.3. Alternative 2.1 was not chosen because it would eliminate an anticipated significant advantage of DB-MHR spent fuel as a permanent waste form. Alternative 2.2 was not chosen because it could result in a significant uncertainty in the calculation of particle integrity, resulting in a less defensible analysis for comparison with repository requirements. Alternative 2.3 was not considered sufficient to eliminate this DDN because of the physical and chemical differences between naturally occurring, bulk SiC deposits and SiC coatings produced by chemical vapor deposition.

4. SCHEDULE REQUIREMENTS

The data must be obtained before final waste package requirements are defined, since long-term particle performance may affect the design of the waste package.

5. PRIORITY

Urgency: TBD
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H

6. CONSEQUENCES OF NONEXECUTION AND FALLBACK POSITION

The consequences of not obtaining the data specified in Section 1.2 would be increased uncertainty in the coating failure projections and increased risk that repository performance assessments based on these projections would be rejected by the regulatory agencies and/or the general public. The fallback position would be to use a combination of Alternatives 2.2 and 2.3. The failure probabilities of the DB-MHR spent fuel coatings for the long times associated with permanent fuel disposal would be estimated based on existing methods and data. To the extent available, geological analogs to SiC coatings would provide insight into their long term strength, so as to provide some confirmation of the coating failure projections. In the limit, complete coating failure and complete kernel dissolution would be assumed, and we would be forced to rely upon retardation and retention by engineered features within the repository (e.g., back-fill) and by the repository geosphere in order to show compliance with radionuclide release and dose regulations.

7. REFERENCES

1. Richards, M. B., and D. W. Ketchen, "PC-MHR Spent Fuel Disposal: Preliminary Evaluation of Whole-Element Disposal Using Multipurpose Canisters," GA/DOE-164-94, General Atomics, September 30, 1994.

2. Miller, G. K., "Determination of Fuel Particle Failure Probabilities and SiC Mean Strengths from Crush Test Data," NPR-MHTGR-0687, EG&G Idaho, September 2, 1993.

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

“AAA” Fuel Program
PyC COATING CORROSION RATES IN AIR
DDN RTW.04.04

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The TRISO coatings on DB-MHR spent fuel particles offer the potential for long-term (e.g., 10^5 years, or more) containment of radionuclides, well beyond that provided by the waste package. Confirmation of the long-term performance of TRISO coatings in a repository environment is needed to support repository performance assessments for comparison with regulatory criteria and to take advantage of the defense-in-depth provided by the multiple release barriers in spent HTGR fuel elements.

The long term, near-field containment of radionuclides by DB-MHR spent fuel depends in large part on the capability of the DF/TF particle coatings to withstand for long times the stresses resulting from internal gas pressure (end-of-irradiation fission gas pressure plus helium pressure due to alpha decay). In the previous PC-MHR spent fuel performance assessment (Ref. 1), it was assumed that the load resulting from the internal gas pressure is borne only by the SiC coating layer; contributions to the coating system structural integrity provided by the outer pyrocarbon (OPyC) coating was neglected. This is conservative, because for a period of time, depending in part on the rate of OPyC corrosion by the repository environment, the OPyC layer will: (1) carry part of the load arising from internal gas pressure and thus reduce the level of stress in the SiC layer, and (2) protect the SiC layer from corrosion from the repository environment. Including these effects will make the DB-MHR spent fuel performance assessment more realistic; however, to do this accurately requires knowledge of the rate of corrosion of the PyC layer in air at expected repository temperatures.

1.1 Summary of Functions/Assumptions

[TBD]

1.2 Current Data Base Summary

There are limited data available which may be used to estimate the corrosion rate of PyC in air at expected repository temperatures. Some measurements have been made of the corrosion of pyrolytic carbon in air at temperatures typical of reactor normal operating and accident conditions which are much higher than those expected to occur in a waste repository (Ref. 2). The PyC corrosion rate determined from these data may be extrapolated to lower temperatures, but the accuracy of this extrapolation is uncertain. Corrosion rates for H-327 and H-451 graphites in air have been measured at temperatures in the range of 375 to 850EC

(Refs. 3 and 4, respectively), which are also well above expected repository temperatures. However, it is not known precisely how to apply these data to estimate pyrolytic carbon corrosion by air at repository temperatures.

1.3 Data Needed

Additional data are needed on the corrosion rate of PyC in air under normal repository conditions and under off-normal conditions, including cycling between wet and dry conditions. With these new data, the contribution of the OPyC layer can be included in the evaluation of the long term integrity of DB-MHR DF/TF particle coatings. This will require accelerated tests of PyC corrosion in dry air and in saturated air conducted at temperatures intermediate between reactor operating temperatures and repository temperatures. The data must be sufficient to enable extrapolation of the corrosion rates to long times (e.g., thousands of years). At least some of the measurements must be made with irradiated TRISO particles to determine whether or not there are irradiation effects.

1.4 Parameters/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications as appropriate:

Test Articles

Fuel particles	TRISO-coated particles in compacts
Matrix type	[Thermosetting resin]
Neutron poisons	PyC-coated B_4C PyC-coated Er_2O_3
Graphite body	Reference fuel-element graphite;[*] unit cell of prismatic element

Normal Repository Environment

Fuel burnup range	[0% to 85]% FIMA - Normal
Peak fast fluence	#8 x 10^{25} n/m^2 (E > 29 fJ)
Repository pressure	1 atm
Residence time	#10^6 years
Fuel element temperatures	[20E to 250EC]

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[*] It is assumed that the reference fuel-element graphite for the DB-MHR will be the functional equivalent of H-451.

Environment	
(Air)	[#0.01 atm H_2O] [0.21 atm O_2] [0.79 atm N_2]
Expected exposed fuel fraction	
Normal conditions	[10^{-5} to 10^{-3}]

Off-Normal Conditions (Groundwater Ingress)

Groundwater chemistry	Yucca Mountain, [TBD}
Vapor-Phase Environment	$O_2/N_2/H_2O/CO/H_2$
Vapor-phase Impurity Levels	[0 to 1.0] atm H_2O
Fuel-element temperatures	[20 - 100 °C]
Groundwater contact time	TBD
Repository pressure	$1 atm
Expected failure fraction	
Off-normal conditions	[10^{-4} to 10^{-2}]

2. DESIGNER'S ALTERNATIVES

Alternatives to the acquisition of the above data are:

2.1 Neglect the contribution provided by the OPyC layer when estimating the failure probability of spent DF/TF coatings.

2.2 Include the contributions of the OPyC layer in the estimate of the failure probability of spent DF/TF coatings by using the limited existing data on PyC and graphite corrosion in air to estimate the corrosion rate of the OPyC by air at repository temperatures.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected design approach is to perform tests to acquire the data specified in Section 1.2. Alternative 2.1 was not chosen because it would result in an overly conservative estimate of the failure probability of the spent fuel particle coatings. Thus, the potential long term benefits of the TRISO coatings for DB-MHR DF/TF disposal would be underestimated. Alternative 2.2 was not chosen because the existing data are insufficient to provide an accurate estimate of OPyC corrosion rate in air at repository conditions.

4. SCHEDULE REQUIREMENTS

The data must be obtained before final waste package requirements are defined, since long-term particle performance may affect the design of the waste package.

5. PRIORITY

Urgency: TBD
Cost benefit: L
Uncertainty in existing data: M
Importance of new data: L

6. CONSEQUENCES OF NONEXECUTION AND FALLBACK POSITION

The consequences of not obtaining the data specified in Section 1.2 would be that the full benefit of the TRISO coatings in retaining radionuclides may not be known. The fallback position would be to use the limited existing data on PyC and graphite corrosion in air and water to estimate the corrosion rate of the OPyC in air. This approach would allow the contribution of the OPyC layer to be included in the calculation of the failure probability of spent DF/TF coatings and would provide a more realistic estimate of the long-term capability of DB-MHR spent fuel particles to retain radionuclides; however, without additional data, the uncertainty in this projection will be large.

7. REFERENCES

1. Richards, M. B. and D. W. Ketchen, "PC-MHR Spent Fuel Disposal: Preliminary Evaluation of Whole-Element Disposal Using Multipurpose Canisters," GA/DOE-164-94, General Atomics, September 30, 1994.

2. Moormann, R., Air Ingress and Graphite Burning in HTRs: A Survey on Analytical Examinations with the Code REACT/THERMIC," Juel-3062, KFA Juelich, May 1995.

3. Jensen, D., et al., "Air/H-327 Graphite Reaction Rate as a Function of Temperature and Irradiation," Gulf-GA-A12647, Gulf General Atomic Company, September 1973.

4. Fuller, E. L., et al., "Corrosion of Nuclear Grade Graphites: Air Oxidation of H-451," ORNL/NPR-91/27, Oak Ridge National Laboratory, October 1992.

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

"AAA" Fuel Program
SiC COATING CORROSION RATES IN AIR
DDN RTW.04.05

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The TRISO coatings on DB-MHR spent fuel particles offer the potential for long-term (e.g., 10^5 years, or more) containment of radionuclides, well beyond that provided by the waste package. Confirmation of the long-term performance of TRISO coatings in a repository environment is needed to support repository performance assessments for comparison with regulatory criteria and to take advantage of the defense-in-depth provided by the multiple release barriers in spent HTGR fuel elements..

The long term, near-field containment of radionuclides by DB-MHR spent fuel depends in part on the capability of the DF/TF SiC coatings to withstand corrosion by air. The previous PC-MHR spent fuel performance assessment (Ref. 1) included an estimate for the expected corrosion of the SiC coatings by air at repository temperatures. This estimate was based on an extrapolation using a parabolic rate law of SiC corrosion rates measured at temperatures much higher than those expected in a repository. However, the accuracy of this estimate is uncertain.

1.1 Summary of Functions/Assumptions

[TBD]

1.2 Current Data Base Summary

There are limited data available which may be used to estimate the long-term corrosion of SiC coatings by air at temperatures expected to occur in a geologic repository. The corrosion rate of SiC in air has been measured at temperatures typical of reactor normal operating and accident conditions (Refs. 2 and 3), which are much higher than those expected to occur in a waste repository. The SiC corrosion rate determined from these data may be extrapolated to lower temperatures using a parabolic rate law, but the accuracy of this extrapolation is uncertain.

1.3 Data Needed

Additional data are needed on the corrosion rates of SiC in air under normal repository conditions and under off-normal conditions, including cycling between wet and dry conditions., to confirm the SiC corrosion kinetics to be used in the DB-MHR spent fuel performance assessment. This will require accelerated tests of SiC corrosion on TRISO particles (from which the OPyC layer has been removed) conducted in dry air and in saturated air and at temperatures intermediate between reactor operating temperatures and repository temperatures. The data must be sufficient to enable extrapolation of the corrosion rates to long times (e.g., hundreds or thousands of years). At least some of the measurements must be made with irradiated TRISO particles with the OPyC layer removed to determine whether or not there are irradiation effects or combined effects of pressure (stress) and chemical reaction.

1.4 Parameters/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications as appropriate:

Test Articles

Fuel particles	TRISO-coated particles in compacts
Matrix type	[Thermosetting resin]
Neutron poisons	PyC-coated B_4C PyC-coated Er_2O_3
Graphite body	Reference fuel-element graphite;* unit cell of prismatic element

Normal Repository Environment

Fuel burnup range	[0% to 85]% FIMA
Peak fast fluence	#8 x 10^{25} n/m^2 (E > 29 fJ)
Repository pressure	1 atm
Residence time	#10^6 years
Fuel element temperatures	[20E to 250EC]
Environment	
(Air)	[#0.01 atm H_2O] [0.21 atm O_2] [0.79 atm N_2]
Expected exposed fuel fraction	
Normal conditions	[10^{-5} to 10^{-3}]

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* It is assumed that the reference fuel-element graphite for the DB-MHR will be the functional equivalent of H-451.

Off-Normal Conditions (Groundwater Ingress)

Groundwater chemistry	Yucca Mountain, [TBD}
Vapor-Phase Environment	$O_2/N_2/H_2O/CO/H_2$
Vapor-phase Impurity Levels	[0 to 1.0] atm H_2O
Fuel-element temperatures	[20 - 100 °C]
Groundwater contact time	TBD
Repository pressure	\$1 atm
Expected failure fraction	
Off-normal conditions	10^{-4} to 10^{-2}]

2. DESIGNER'S ALTERNATIVES

Alternatives to the acquisition of the above data are:

2.1 Use the limited existing data on SiC corrosion in air at high temperatures to estimate the corrosion rate of the SiC in air at repository temperatures.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected design approach is to perform tests to acquire the data specified in Section 1.2. Alternative 2.1 was not chosen because use of the existing data requires a large extrapolation to repository conditions which entails large, but poorly quantified, uncertainties.

4. SCHEDULE REQUIREMENTS

The data must be obtained before final waste package requirements are defined, since long-term particle performance may affect the design of the waste package.

5. PRIORITY

Urgency: TBD
Cost benefit: M
Uncertainty in existing data: M
Importance of new data: M

6. CONSEQUENCES OF NONEXECUTION AND FALLBACK POSITION

The consequences of not obtaining the data specified in Section 1.2 would be that the accuracy of the estimate for the corrosion by air of the SiC coatings, and consequently the accuracy of the spent fuel performance assessment, would be highly uncertain. The fallback position would be to use the limited existing data on SiC corrosion in air at high temperatures to estimate the corrosion rate of the SiC in air at repository temperatures. However, without additional data, the uncertainty in the long term spent fuel performance will be difficult to estimate.

7. REFERENCES

1. Richards, M. B., and D. W. Ketchen, "PC-MHR Spent Fuel Disposal: Preliminary Evaluation of Whole-Element Disposal Using Multipurpose Canisters," GA/DOE-164-94, General Atomics, September 30, 1994.

2. Shiroky, G. H., et al., "Oxidation Characteristics of CVD Silicon Carbide and Silicon Nitride," GA-A18696, GA Technologies, December 1986.

3. Vaughn, W. L., and H. G. Maahs, "Active to Passive Transition in the Oxidation of Silicon Carbide and Silicon Nitride in Air," J. Am. Ceram. Soc., **73**, 1540 (1990).

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

“AAA” Fuel Program
RESIN MATRIX CORROSION RATES IN AIR
DDN RTW.04.06

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The long term, near-field containment of radionuclides by DB-MHR spent fuel depends in large part on the capability of the fuel particle coatings to withstand for long times the stresses resulting from internal gas pressure (end-of-irradiation fission gas pressure plus helium pressure due to alpha decay. It is believed that the compact thermosetting resin matrix will contribute to the long-term radionuclide retention of the fuel, by providing a fuel form which is mechanically stable for long times. For this to be evaluated requires knowledge of the long-term rate of corrosion by air of DB-MHR resin matrix at temperatures expected in a waste repository.

The resin matrix is also a potential source of ^{14}C release to the environment. An accurate estimate of the long-term rate of corrosion by air of DB-MHR resin matrix at temperatures expected in a waste repository is also required to asses potential ^{14}C release.

1.1 Summary of Functions/Assumptions

[TBD]

1.2 Current Data Base Summary

There are limited data available which may be used to estimate the rate of corrosion by air of resin matrix at repository temperatures. Some measurements have been made of the corrosion of resin matrix in air, including data from the German HTR program, but these measurements have been for temperatures representative of reactor normal operation and accident conditions (Ref. 1), which are well above the temperatures expected in a waste repository. Corrosion rates for H-327 and H-451 graphites by air have been measured at temperatures in the range of 375 to 850EC (Refs. 2 and 3), which are also well above anticipated repository temperatures. These data have been applied to estimate the corrosion rate of resin matrix by air at repository temperatures by using an Arrhenius equation to extrapolate to lower temperature and assuming a factor of twenty higher rate of corrosion for resin matrix (compared to H-451 graphite). This assumption is reasonable for petroleum pitch matrix which has high interconnected porosity and relatively high metallic impurities, which may serve as catalysts, but it is probably excessively conservative for denser and purer resin matrix materials.

1.3 Data Needed

Additional data are needed on the corrosion rate of reference resin matrix in air under normal repository conditions and under off-normal conditions, including cycling between wet and dry conditions. With these new data, the matrix can be included in the evaluation of the long term capability of DB-MHR spent fuel to retain radionuclides. Generating these data will require accelerated tests of resin matrix corrosion in dry air and in saturated air conducted at temperatures intermediate between reactor operating temperatures and the expected repository temperatures (i.e., the reaction rates must measurable over practical time frames). The data must be sufficient to enable extrapolation of the corrosion rates to long times (e.g., thousands of years). At least some of the measurements must be made with irradiated matrix recovered from irradiated fuel compacts to determine whether or not there are irradiation effects, including possible catalysis by metallic fission products sorbed on the matrix.

1.4 Parameters/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications as appropriate:

Test Articles

Matrix type	[Thermosetting resin]

Normal Repository Environment

Peak fast fluence	#8 x 10^{25} n/m^2 (E > 29 fJ)
Repository pressure	1 atm
Residence time	#10^6 years
Fuel element temperatures	[20E to 250EC]
Environment (Air)	[#0.01 atm H_2O] [0.21 atm O_2] [0.79 atm N_2]
Expected exposed fuel fraction Normal conditions	[10^{-5} to 10^{-3}]

Off-Normal Conditions (Groundwater Ingress)

Groundwater chemistry	Yucca Mountain, [TBD}

Vapor-Phase Environment	$O_2/N_2/H_2O/CO/H_2$
Vapor-phase Impurity Levels	[0 to 1.0] atm H_2O
Fuel-element temperatures	[20 - 100 ^oC]
Groundwater contact time	TBD
Repository pressure	$1 atm
Expected failure fraction Off-normal conditions	[10^{-4} to 10^{-2}]

2. DESIGNER'S ALTERNATIVES

Alternatives to the acquisition of the above data are:

2.1 Neglect the contribution provided by the resin matrix when estimating the long-term capability of DB-MHR spent fuel to retain radionuclides.

2.2 Include the contribution of the resin matrix in the estimate of the long-term capability of DB-MHR spent fuel to retain radionuclides by using the limited existing data on resin-matrix corrosion in air, particularly the German data, to estimate the corrosion rate of resin matrix in air at repository temperatures.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected design approach is to perform tests to acquire the data specified in Section 1.2. This will allow the contribution made by the resin matrix in the long-term radionuclide retention by DB-MHR fuel to be evaluated. Alternative 2.1 was not chosen because it would result in underestimating the potential long-term capability of DB-MHR spent fuel to retain radionuclides. Alternative 2.2 was not chosen because the existing data are insufficient to provide an accurate estimate of resin matrix corrosion rate in air at repository conditions.

4. SCHEDULE REQUIREMENTS

The data must be obtained before final waste package requirements are defined, since long-term particle performance may affect the design of the waste package.

5. PRIORITY

Urgency: TBD
Cost benefit: L
Uncertainty in existing data: M
Importance of new data: L

6. CONSEQUENCES OF NONEXECUTION AND FALLBACK POSITION

The consequences of not obtaining the data specified in Section 1.2 would be that the full potential of DB-MHR fuel in retaining radionuclides may not be known; the degree of conservatism is not expected to be large. The fallback position would be to use the limited existing data on resin-matrix and graphite corrosion in air at high temperatures to estimate the corrosion rate of the resin matrix in air at repository temperatures. This would allow the contribution of the resin matrix to be included in the estimate of the long-term capability of DB-MHR spent fuel particles to retain radionuclides. However, without additional data, the uncertainty in this projection will be unknown.

7. REFERENCES

1. Moormann, R., “Air Ingress and Graphite Burning in HTRs: A Survey on Analytical Examinations with the Code REACT/THERMIC,” Juel-3062, KFA Juelich, May 1995.

2. Jensen, D., et al., "Air/H-327 Graphite Reaction Rate as a Function of Temperature and Irradiation," Gulf-GA-A12647, Gulf General Atomic Company, September 1973.

3. Fuller, E. L., et al., "Corrosion of Nuclear Grade Graphites: Air Oxidation of H-451," ORNL/NPR-91/27, Oak Ridge National Laboratory, October 1992.

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

“AAA” Fuel Program
GRAPHITE CORROSION RATES IN AIR
DDN RTW.04.07

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The long term, near-field containment of radionuclides by DB-MHR spent fuel depends in large part on the capability of the fuel particle coatings to withstand for long times the stresses resulting from internal gas pressure (end-of-irradiation fission gas and CO pressure plus helium pressure due to alpha decay). The fuel element body, which is made from high-purity, near-isotropic nuclear graphite,[*] will also contribute to the long-term radionuclide retention of the fuel by providing a fuel form which is mechanically stable for long times and by serving as a sorptive sink for condensable radionuclides released from the fuel compacts in the reactor and in the repository. For this to be evaluated requires knowledge of the long-term rate of corrosion by air of reference DB-MHR graphite at temperatures expected in a geologic repository.

The fuel-element graphite is also a potential source of release of ^{14}C and other sorbed radionuclides to the environment. An accurate estimate of the long-term rate of corrosion by air of reference DB-MHR graphite at temperatures expected in a waste repository is also required to accurately assess potential ^{14}C release, which is a dominant contributor to estimated offsite doses from a conceptual geologic repository containing spent DB-MHR fuel elements.

1.1 Summary of Functions/Assumptions

[TBD]

1.2 Current Data Base Summary

There are limited data available which may be used to estimate the rate of corrosion by air of H-451 graphite at repository temperatures. Corrosion rates for H-327 and H-451 graphites by air have been measured at temperatures in the range of 375 to 850EC (Refs. 1 and 2), which are well above anticipated repository temperatures. These data have been applied to estimate the corrosion rate of H-451 graphite by air at repository temperatures by using an Arrhenius equation to extrapolate to lower temperature. However, the accuracy of this estimate is not known.

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[*] It is assumed that the reference fuel-element graphite for the DB-MHR will be the functional equivalent of H-451.

1.3 Data Needed

Additional data are needed to allow a more accurate estimate of the corrosion rate of reference DB-MHR graphite in air at repository temperatures. This will require accelerated tests of reference graphite corrosion in air conducted at temperatures near those expected in the repository. The data must be sufficient to enable extrapolation of the corrosion rates to long times (e.g., hundreds or thousands of years). At least some of the measurements must be made with irradiated graphite to determine whether or not there are irradiation effects, including possible catalysis by metallic fission products sorbed on the graphite.

1.4 Parameters/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications as appropriate:

Test Articles

Graphite body	Reference fuel-element graphite; unit cell of prismatic element

Normal Repository Environment

Peak fast fluence	#8 x 10^{25} n/m^2 (E > 29 fJ)
Repository pressure	1 atm
Residence time	#10^6 years
Fuel element temperatures	[20 to 250EC]
Environment (Air)	[#0.01 atm H_2O] [0.21 atm O_2] [0.79 atm N_2]
Expected exposed fuel fraction Normal conditions	[10^{-5} to 10^{-3}]

Off-Normal Conditions (Groundwater Ingress)

Groundwater chemistry	Yucca Mountain, [TBD}
Vapor-Phase Environment	$O_2/N_2/H_2O/CO/H_2$
Vapor-phase Impurity Levels	[0 to 1.0] atm H_2O

Fuel-element temperatures	[20 - 100 °C]
Groundwater contact time	TBD
Repository pressure	≥1 atm
Expected failure fraction	
Off-normal conditions	[10^{-4} to 10^{-2}]

2. DESIGNER'S ALTERNATIVES

Alternatives to the acquisition of the above data are:

2.1 Use the existing high temperature graphite air corrosion data to estimate the long-term integrity of the DB-MHR spent fuel element as well as to estimate the expected release as $^{14}CO_2$.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected approach is to perform tests to acquire the data specified in Section 1.2. This approach will allow the contribution made by the fuel-element graphite in the long-term radionuclide retention by DB-MHR fuel to be evaluated, as well as improve the accuracy of the estimate of ^{14}C release. Alternative 2.1 was not chosen because it would result in an uncertain estimate for the long-term stability of the spent fuel element, as well as an uncertain estimate for ^{14}C release.

4. SCHEDULE REQUIREMENTS

The data must be obtained before final waste package requirements are defined, since long-term particle performance may affect the design of the waste package.

5. PRIORITY

Urgency: TBD
Cost benefit: M
Uncertainty in existing data: M
Importance of new data: M

6. CONSEQUENCES OF NONEXECUTION AND FALLBACK POSITION

The consequences of not obtaining the data specified in section 1.2 would be that the estimate for the long-term stability of the DB-MHR spent fuel element and the estimate of ^{14}C release would be uncertain. The fallback position would be to use the limited existing data on graphite corrosion in air at high temperatures to estimate the corrosion rate of the graphite in air at repository temperatures. This would allow the long-term stability of the graphite fuel element and the expected ^{14}C release to be estimated.

However, without additional data, the attendant uncertainty in the long-term spent fuel performance will be difficult to estimate.

7. REFERENCES

1. Jensen, D., et al., "Air/H-327 Graphite Reaction Rate as a Function of Temperature and Irradiation," Gulf-GA-A12647, Gulf General Atomic Company, September 1973.
2. Fuller, E. L., et al., "Corrosion of Nuclear Grade Graphites: Air Oxidation of H-451," ORNL/NPR-91/27, Oak Ridge National Laboratory, October 1992.

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

"AAA" Fuel Program
GRAPHITE NONCOMBUSTIBILITY DEMONSTRATION
DDN RTW.04.08

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

A generic requirement (given in 10CFR60) for high-level waste disposal in a geologic repository is that the waste form must be "noncombustible" (the proposed Yucca Mountain site-specific criteria, to be promulgated as 10CFR63, do not currently include such a requirement). No specific definition of combustibility is provided in 10CFR60; however, a common standard used to assess the combustibility of materials is the ASTM E-136 standard. The available (albeit circumstantial) evidence indicates that H-451 - and, by inference, the reference DB-MHR graphite[1] - is clearly □noncombustible□ when judged by this ASTM standard.

1.1 Summary of Functions/Assumptions

[TBD]

1.2 Current Data Base Summary

As stated above, a common standard used to assess the combustibility of materials is ASTM E-136 "Standard Test Method for Behavior of Materials in a Vertical Tube Furnace at 750EC." H-451 has not been tested to this standard; however, a different nuclear-grade graphite, designated TSX, has been tested (Ref. 1). No flaming was observed from any of the TSX test specimens, which is one of the ASTM E-136 criteria for noncombustibility. However, the temperature rise in the specimens slightly exceeded the 30EC temperature rise allowed by ASTM E-136[2] (Three of four specimens showed a temperature rise of 32EC, and the fourth showed a temperature rise of 35EC).

An evaluation of the expected performance of H-451 graphite in the ASTM E-136 test based on the relative properties of H-451 and TSX has indicated that H-451 would pass the test (i.e., would be rated as noncombustible). This evaluation is supported by the results of H-451 graphite oxidation tests performed at LANL to validate safety methods. In those tests H-451 graphite tubes simulating the geometry of a single coolant channel of an MHR fuel element were heated in a flowing oxygen/nitrogen mixture corresponding to conditions which might occur

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[1] It is assumed that the reference fuel-element graphite for the DB-MHR will be the functional equivalent of H-451.

[2] It is noteworthy that an analogous international standard for combustibility, ISO R 1182, allows a temperature rise of up to 50 °C.

in an MHR core during an air ingress accident. Even after heating at 700EC for 6.5 hours, the measured graphite temperature rise was only 25EC (Ref. 2).

1.3 Data Needed

Additional data are needed to confirm the initial evaluation that reference DB-MHR graphite is noncombustible according to the criteria outlined in ASTM standard E-136. The most direct way to provide this confirmation would be to test graphite specimens in accordance with the ASTM standard. Both unirradiated and irradiated specimens should be tested. Additional testing may ultimately be required if regulatory authorities were to adopt a different quantitative definition of combustibility for the repository.

1.4 Parameters/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications as appropriate:

Test Articles

Graphite body	Reference fuel-element graphite; unit cell of prismatic element

Normal Repository Environment

Peak fast fluence	#8 x 10^{25} n/m^2 (E > 29 fJ)
Repository pressure	1 atm
Residence time	#10^6 years
Fuel element temperatures	[20E to 250EC]
Environment (Air)	[#0.01 atm H_2O] [0.21 atm O_2] [0.79 atm N_2]

Off-Normal Conditions (Groundwater Ingress)

Groundwater chemistry	Yucca Mountain, [TBD}
Vapor-Phase Environment	$O_2/N_2/H_2O/CO/H_2$
Vapor-phase Impurity Levels	[0 to 1.0] atm H_2O
Fuel-element temperatures	[20 - 100 ^oC]

Groundwater contact time	TBD
Repository pressure	≥1 atm

2. DESIGNER'S ALTERNATIVES

Alternatives to the acquisition of the above data are:

2.1 Continue to evaluate reference graphite noncombustibility based on analytical assessments of the existing ASTM E-136 test data for TSX graphite and the existing LANL H-451 oxidation data .

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected design approach is to perform tests to acquire the data specified in Section 1.2. This will allow a direct confirmation of the noncombustibility of reference DB-MHR graphite vis-à-vis the ASTM E-136 combustibility standard. Alternative 2.1 was not chosen because it is only circumstantial evidence of the noncombustibility of H-451 graphite. Because of the Windscale and Chernobyl reactor accidents, the question of graphite “combustibility” has had considerable visibility, and it would likely be a licensing issue for whole-element disposal in a geologic repository. Consequently, it is very important to reach early agreement with the regulatory authorities on a quantitative definition of combustibility and then to demonstrate unequivocally by an appropriate test program that the reference DB-MHR graphite is “noncombustible” by that definition.

4. SCHEDULE REQUIREMENTS

The data must be obtained before final waste package requirements are defined, since long-term particle performance may affect the design of the waste package.

5. PRIORITY

Urgency: TBD
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H

6. CONSEQUENCES OF NONEXECUTION AND FALLBACK POSITION

The consequences of not obtaining the data specified in Section 1.2 would be that the determination of the reference DB-MHR graphite noncombustibility, and thus compliance with one of the generic requirements for waste disposal in a geologic repository, would be to introduce significant licensing uncertainty. The fallback position would be to continue to evaluate graphite noncombustibility based on analytical assessments of the existing ASTM E-136 test data for TSX graphite and the existing LANL H-451 oxidation data. This fallback position appears to embody unacceptable programmatic risk and is not recommended, especially considering that such test are typically straightforward and inexpensive.

7. REFERENCES

1. Quapp, W. J., et al., "A Post-Chernobyl Review of Nuclear Graphite Oxidation and Combustion," Presented at the Eighth Annual International HTGR Conference, September 15-16, 1986, San Diego, CA.

2. Richards, M. B., and D. W. Ketchen, "PC-MHR Spent Fuel Disposal: Preliminary Evaluation of Whole-Element Disposal Using Multipurpose Canisters," GA/DOE-164-94, General Atomics, September 30, 1994.

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

"AAA" Fuel Program
PyC COATING CORROSION RATES (GROUNDWATER)
DDN RTW.04.09

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The long-term containment of spent TRISO-coated fuel will depend largely on the performance of the pyrolytic carbon (PyC) and silicon carbide (SiC) coatings. The PyC coatings may be corroded by groundwater during off-normal repository environment conditions (i.e., groundwater ingress). Corrosion of the outer PyC coating is a potentially important mechanism that can affect long-term fuel performance. Therefore, corrosion characteristics of the outer PyC (OPyC) coating must be quantified for the off-normal conditions.

1.1 Summary of Functions/Assumptions

[TBD]

1.2 Current Data Base Summary

Studies were performed at Oak Ridge National Laboratory (ORNL) on simulated high-level nuclear waste coated with PyC and SiC in order to improve waste containment relative to glassified waste forms (Ref. 1). Using the technology developed for manufacturing HTGR coated fuel particles, microspheres of simulated Savannah River Plant waste were coated with PyC and SiC coatings to reduce leachability. Leach tests on PyC-coated samples were run at 90°C for times up to 28 days. The analysis found no detectable release of Cs, U, Sr, Zr, etc., whereas measurable leach rates have been obtained for glassified waste forms under these conditions.

The corrosion behavior of graphite, PyC and SiC has also been investigated for the purpose of evaluating improved barriers for nuclear-waste isolation (Refs. 2 and 3). Measurements were made of graphite, PyC and SiC corrosion in deionized water at 200 to 300°C in a pressurized autoclave. Measurements of CO_2 and CO in water were used to quantify the corrosion rates. The results indicated that the leach rate of graphite in deionized water was more than a factor of 10^5 slower than the leach rate of waste glass. The corrosion rates of PyC and SiC were not significantly different, but were apparently 5 to 10x higher than the graphite corrosion rates (an exact ranking is difficult since the reported corrosion rates were not normalized by the BET surface areas). Applying the results of this test to a PC-MHR TRISO-coated particle at a temperature of 96°C (the boiling point of water at Yucca Mountain elevation), it was estimated that it would take 10^5 to 10^6 years to completely dissolve the 40 μm outer PyC coating (Ref. 4).

1.3 Data Needed

Correlations are needed describing the corrosion rates for the OPyC coatings of irradiated, TRISO-coated fuel particles during off-normal (groundwater ingress) conditions, including cycling between wet and dry conditions. High-burnup Pu particles recovered from fuel test element FTE-13 irradiated in the Peach Bottom-1 reactor, which is currently in long-term storage, would be an ideal source of test specimens. Data are needed from accelerated tests with a range of fuel temperatures for normal and off-normal conditions and a range of groundwater chemistries.

1.4 Parameter/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications as appropriate:

Test Articles

Fuel particles	TRISO-coated particles in compacts
Matrix type	[Thermosetting resin]
Neutron poisons	PyC-coated B_4C PyC-coated Er_2O_3
Graphite body	Reference fuel-element graphite;[*] unit cell of prismatic element

Normal Repository Environment

Fuel burnup range	[0 to 85 % FIMA]
Peak fast fluence	[$\geq 8 \times 10^{25}$ n/m^2 (E > 29 fJ)]
Repository pressure	1 atm
Residence time	$\leq 10^6$ years
Fuel element temperatures	[20^o to 250^{o}C]
Environment (Air)	[$\leq$0.01 atm H_2O] [0.21 atm O_2] [0.79 atm N_2]

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[*] It is assumed that the reference fuel-element graphite for the DB-MHR will be the functional equivalent of H-451.

Expected exposed fuel fraction Normal conditions	[10^{-5} to 10^{-3}]

Off-Normal Conditions (Groundwater Ingress)

Groundwater chemistry	Yucca Mountain, [TBD}
Vapor-phase environment	$O_2/N_2/H_2O/CO/H_2$
Vapor-phase impurity levels	[0 to 1.0] atm H_2O
Fuel-element temperatures	[20 - 100°C]
Groundwater contact time	TBD
Repository pressure	≥ 1 atm
Expected failure fraction Off-normal conditions	[10^{-4} to 10^{-2}]

2. DESIGNER'S ALTERNATIVES

Alternatives to the acquisition of the above data are:

2.1 Use the existing database and add design margins to account for the uncertainties.

2.2 Do not take credit for the OPyC coating when evaluating in-repository fuel performance.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Measure the corrosion rates for the OPyC coating of the irradiated TRISO-coated fuel in groundwater in accelerated tests. The corrosion rates will be measured as a function of groundwater chemistry, fuel temperature, and time.

Alternative 2.1 is rejected because the extrapolation of the existing data base for significantly larger times and for different groundwater chemistries and temperatures will introduce large uncertainties. The ORNL test duration was 28 days, while the other tests were run for 24 hours, and these tests were done with deionized water. The analysis of these test results will give average rates for the those time periods, but there is no way to differentiate initial rates from long-term rates; extrapolation in time over several orders of magnitude is highly uncertain. Also, there is uncertainty in the data base, as the three pyrocarbon samples exhibited unexplained corrosion rate variations. Alternative 2.2 is rejected as being excessively conservative; it is expected that the OPyC will provide significant corrosion resistance, thereby protecting the SiC coating, and will contribute to the retention of radionuclides, thereby providing defense-in-depth.

4. SCHEDULE REQUIREMENTS

TBD

5. PRIORITY

Urgency: TBD
Cost benefit: M
Uncertainty in existing data: M
Importance of new data: M

6. CONSEQUENCES OF NONEXECUTION AND FALLBACK POSITION

The fallback position would be to use Alternative 2.1, which will lead to large uncertainties and conservative estimates of OPyC corrosion. A weakened position during safety review would result from this uncertainty, and reviewers may not give credit for the OPyC as part of the multi-shelled pressure vessel or as a radionuclide retention barrier.

7. REFERENCES

1. Stinton, D. P., et al., "Coating of Crystalline Nuclear Waste Forms to Improve Inertness," Proceedings of the 83rd Annual Meeting, The American Ceramic Society, Washington, DC, May 6, 1981, pp. 394-398.

2. Gray, W. J., "Reaction of Graphite with Water and its Implications for Radioactive Waste Storage," Radioactive Waste Management, Vol. 1(1), May 1980, pp. 105-109.

3. Gray, W. J., "A study of the Oxidation of Graphite in Liquid Water for Radioactive Storage Applications," Radioactive Waste Management and the Nuclear Fuel Cycle, Vol. 3(2), December 1982, pp. 137-149.

4. Richards, M. B., and D. W. Ketchen, "PC-MHR Spent Fuel Disposal: Preliminary Evaluation of Whole-Element Disposal Using Multipurpose Canisters," GA/DOE-164-94, General Atomics, September 30, 1994.

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

“AAA” Fuel Program
SiC COATING DISSOLUTION RATES (GROUNDWATER)
DDN RTW.04.10

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The long-term containment of spent TRISO coated fuel will depend largely on the performance of the pyrolytic carbon (PyC) and silicon carbide (SiC) coatings. Following the dissolution of the outer PyC (OPyC) coating by groundwater interaction, the SiC coatings may be corroded by groundwater during off-normal repository environment conditions (i.e., groundwater ingress). Corrosion of the SiC is potentially the controlling mechanism in determining long-term fuel performance. Therefore, corrosion characteristics of the SiC coating must be quantified for the off-normal conditions.

1.1 Summary of Functions/Assumptions

[TBD]

1.2 Current Data Base Summary

Studies were performed at Oak Ridge National Laboratory (ORNL) on simulated high-level nuclear waste coated with PyC and SiC in order to improve waste containment relative to glassified waste forms (Ref. 1). Using the technology developed for manufacturing HTGR coated fuel particles, microspheres of simulated Savannah River Plant waste were coated with PyC and SiC coatings to reduce leachability. Leach tests on PyC-coated samples ran at 90 ^{o}C for times up to 28 days. The analysis found no detectable release of Cs, U, Sr, Zr, etc., whereas measurable leach rates have been obtained for glassified waste forms under these conditions. Coated particle waste was especially effective for immobilizing Cs, which readily leached from glassified waste because of its high solubility.

Experiments were also performed on the corrosion behavior of graphite, PyC and SiC in order to evaluate improved barriers for nuclear-waste isolation (Refs. 2 and 3). Measurements were made of graphite, PyC and SiC corrosion in deionized water at 200 to 300 ^{o}C in a pressurized autoclave. Measurements of CO_2 and CO in water were used to quantify the corrosion rates. The tests were run for 24 hours and then the gases were analyzed. The analysis of the data showed that the corrosion rates of PyC and SiC were not significantly different, but were 5 to 10x higher than the graphite corrosion rates (an exact ranking is difficult since the reported corrosion rates were not normalized by the BET surface areas). The leach rate of graphite in deionized water was more than a factor of 10^5 slower than the leach rate of waste glass. Applying the SiC corrosion rate, calculated from these test data, to a PC-MHR TRISO-coated particle a temperature of 96°C (the boiling point of water at Yucca Mountain elevation), it was estimated that it would take 3×10^6 years to completely dissolve the 35 μm SiC coating after the OPyC coating has failed (Ref. 4).

1.3 Data Needed

Correlations are needed describing the corrosion rates for the SiC coatings of irradiated TRISO-coated particles during the off-normal (groundwater ingress) conditions, including cycling between wet and dry conditions. High-burnup Pu particles recovered from fuel test element FTE-13 irradiated in the Peach Bottom-1 reactor, which is currently in long-term storage, would be an ideal source of test specimens. Data are needed from the accelerated tests with a range of water chemistries and fuel temperatures.

1.4 Parameter/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications as appropriate:

Test Articles

Fuel particles	TRISO-coated particles in fuel compacts
Matrix type	[Thermosetting resin]
Neutron poisons	PyC-coated B_4C PyC-coated Er_2O_3
Graphite body	Reference fuel-element graphite;* unit cell of prismatic element

Normal Repository Environment

Fuel burnup range	[0 to 85]% FIMA
Peak fast fluence	[$\geq 8 \times 10^{25}$] n/m^2 (E > 29 fJ)
Repository pressure	1 atm
Residence time	$\leq 10^6$ years
Fuel element temperatures	[20 to 250 °C]

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[*] It is assumed that the reference fuel-element graphite for the DB-MHR will be the functional equivalent of H-451.

Environment	
(Air)	[≤ 0.01 atm H_2O]
	[0.21 atm O_2]
	[0.79 atm N_2]
Expected exposed fuel fraction	
Normal conditions	[10^{-5} to 10^{-3}]

Off-Normal Conditions (Groundwater Ingress)

Groundwater chemistry	Yucca Mountain, [TBD}
Vapor-phase environment	$O_2/N_2/H_2O/CO/H_2$
Vapor-phase impurity levels	[0 to 1.0] atm H_2O
Fuel-element temperatures	[20 – 100 °C]
Groundwater contact time	TBD
Repository pressure	≥ 1 atm
Expected failure fraction	
Off-normal conditions	[10^{-4} to 10^{-2}]

2. DESIGNER'S ALTERNATIVES

Alternatives to the acquisition of the above data are:

2.1 Use the existing data base and add design margins to account for the uncertainties.

2.2 Assume that SiC coatings fail upon contact with groundwater and rely upon other engineered barriers and the repository geosphere to retard and retain radionuclides.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Measure the corrosion rates for the SiC coating of spent DB-MHR fuel in groundwater in accelerated tests. The corrosion rates will be measured as a function of groundwater chemistry, and fuel temperature.

Alternative 2.1 is rejected because the extrapolation of the existing data base for significantly larger times and for different groundwater chemistries and fuel temperatures will introduce large uncertainties. The tests from which the SiC corrosion rates were calculated were run for 24 hours. The analysis of these test results will give average rates for that time period, but there is no way to differentiate initial rates from long-term rates;

extrapolation in time over several orders of magnitude will introduce large uncertainties. Also, extrapolating the corrosion rates to lower temperatures than the test temperature of 200 to 300°C involve large rate changes with further uncertainties. Alternative 2.2 is rejected because it would lead to gross overprediction of source terms and off-site dose rates.

4. SCHEDULE REQUIREMENTS

TBD

5. PRIORITY

Urgency: TBD
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H

6. CONSEQUENCES OF NONEXECUTION AND FALLBACK POSITION

The fallback position would be to use Alternative 2.1, which will lead to conservative estimates of SiC corrosion. A weakened position during safety review would result from this uncertainty; ultimately, reviewers might conservatively assume that the SiC fails when contacted by groundwater (i.e., Alternative 2.2). Under these circumstances, the expected advantages of spent DB-MHR fuel elements as permanent waste form compared to spent LWR fuel elements would be lost.

7. REFERENCES

1. Stinton, D. P., et al., "Coating of Crystalline Nuclear Waste Forms to Improve Inertness," Proceedings of the 83rd Annual Meeting, The American Ceramic Society, Washington, DC, May 6, 1981, pp. 394-398.

2. Gray, W. J., "Reaction of Graphite with Water and its Implications for Radioactive Waste Storage," Radioactive Waste Management, Vol. 1(1), May 1980, pp. 105-109.

3. Gray, W. J., "A study of the Oxidation of Graphite in Liquid Water for Radioactive Storage Applications," Radioactive Waste Management and the Nuclear Fuel Cycle, Vol. 3(2), December 1982, pp. 137-149.

4. Richards, M. B., and D. W. Ketchen, "PC-MHR Spent Fuel Disposal: Preliminary Evaluation of Whole-Element Disposal Using Multipurpose Canisters," GA/DOE-164-94, General Atomics, September 30, 1994.

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

“AAA” Fuel Program
RESIN-MATRIX CORROSION RATES (GROUNDWATER)
DDN RTW.04.11

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The carbonaceous matrix materials used as binders in the fuel compacts, lumped burnable-poison (LBP) compacts, and the reserve shutdown control (RSC) pellets consist of finely divided graphite flakes bonded together with residual carbon from carbonized, petroleum-pitch or thermosetting resin binders. The matrix may be corroded by groundwater during off-normal repository environment conditions (i.e., groundwater ingress). The corrosion characteristics of the DB-MHR reference resin-matrix materials must be quantified for the groundwater ingress conditions.

1.1 Summary of Functions/Assumptions

[TBD]

1.2 Current Data Base Summary

The corrosion rates of petroleum pitch-derived compact matrix in water are assumed to be functionally similar to those for the H-451 graphite, except that the rate is assumed to be 10x higher. The reaction of water with H-451 graphite exhibits Langmuir-Hinshelwood type kinetics with significant product inhibition by H_2 but not by CO. Corrosion rates of German thermosetting resin-derived matrix materials at the elevated temperatures characteristic of in-reactor service have been measured at KFA (e.g., Ref. 1), but the data have not been evaluated by the DB-MHR program. In principle, thermosetting resin-derived matrix materials would be expected to be less permeable and less reactive with water than pitch-derived matrix materials because the latter are characterized by lower densities, higher interconnected porosities, and much higher residual impurities, including catalytically active transition metals. Long-term leaching tests with irradiated AVR fuel spheres in brine have been on-going at KFA for >15 years; no significant dissolution of the sphere matrix has been reported (Ref. 2).

1.3 Data Needed

Correlations are needed describing the corrosion rates for resin-matrix materials during the off-normal (groundwater ingress) conditions, including cycling between wet and dry conditions. To characterize the reaction kinetics, the reaction rate must be determined as a function of groundwater chemistry, temperature, pressure, and time. The effect of partial matrix corrosion on both the mass transfer processes and the reaction kinetics must also be quantified.

1.4 Parameter/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications as appropriate:

Test Articles

Matrix type	[Thermosetting resin]
Neutron poisons	PyC-coated B_4C PyC-coated Er_2O_3
Graphite body	Reference fuel-element graphite;[*] unit cell of prismatic element

Normal Repository Environment

Fuel burnup range	[0 to 85]% FIMA - Normal
Peak fast fluence	[$\leq 8 \times 10^{25}$] n/m^2 (E > 29 fJ)
Repository pressure	1 atm
Residence time	$\leq 10^6$ years
Fuel element temperatures	[20 to 250 ^{o}C]
Environment (Air)	[$\leq$ 0.01 atm H_2O] [0.21 atm O_2] [0.79 atm N_2]

Off-Normal Conditions (Groundwater Ingress)

Groundwater chemistry	Yucca Mountain, [TBD}
Vapor-phase environment	$O_2/N_2/H_2O/CO/H_2$
Vapor-phase impurity levels	[0 to 1.0] atm H_2O
Fuel-element temperatures	[20 – 100 ^{o}C]
Groundwater contact time	TBD
Repository pressure	$\geq$ 1 atm

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[*] It is assumed that the reference fuel-element graphite for the DB-MHR will be the functional equivalent of H-451.

2. DESIGNER'S ALTERNATIVES

Alternatives to the acquisition of the above data are:

2.1 Use the existing database for graphite corrosion rates in water and assume that the resin-matrix corrosion rates are 10x higher. Add design margins to account for the uncertainties.

2.2 Assume that the compact matrix is completely dissolved upon contact with groundwater and rely upon the OPyC and SiC coatings and graphite for radionuclide retention.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Perform accelerated tests in order to measure the corrosion rates for the reference resin-matrix material in groundwater. The corrosion rates will be measured as a function of groundwater chemistry, temperature, system pressure, and time. Alternative 2.1 is rejected because the use of the existing data base and the assumptions will lead to large uncertainties and will require excessive design margins. Alternative 2.2 is rejected as excessively conservative and contrary to the available German data.

4. SCHEDULE REQUIREMENTS

TBD

5. PRIORITY

Urgency: TBD
Cost benefit: M
Uncertainty in existing data: M
Importance of new data: M

6. CONSEQUENCES OF NONEXECUTION AND FALLBACK POSITION

The fallback position would be to use Alternative 2.1, which will lead to large uncertainties and conservative estimates of resin-matrix corrosion rates. A weakened position during safety review would result from this uncertainty; ultimately, reviewers might also conservatively assume that the matrix completely dissolves when contacted by groundwater (i.e., Alternative 2.2).

7. REFERENCES

1. Loenissen, K.-J., et al., “Graphitkorrosion durch Wasserdampf in HTRs bei Stoerfaellen mit hohen Systemdruck,” (Proc. Jahrestagung Kerntechnik ’87, Muenchen, 1987), Inforum GmbH, 1987, pp. 171-174.

2. Kirch, N., U. Brinkmann, and H. Bruecher, "Storage and Disposal of HTR Fuel in FRG," Nucl. Eng. Des., **121**, 241 (1990).

Originator	Date
Engineering Manager	Date
Project Manager	Date

DATE: 09/27/02

"AAA" Fuel Program
GRAPHITE CORROSION RATES (GROUNDWATER)
DDN RTW.04.12

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The DB-MHR fuel compacts are contained in high purity, nuclear graphite fuel elements.* During postulated off-normal repository environment conditions of groundwater ingress, the fuel element graphite may be corroded by groundwater. Corrosion of the fuel element graphite is a potentially important mechanism that can affect long-term fuel performance in the repository environment, including the mechanical stability of the waste form. Therefore, corrosion characteristics of the reference fuel-element graphite must be quantified for the groundwater ingress conditions.

1.1 Summary of Functions/Assumptions

[TBD]

1.2 Current Data Base Summary

Experiments have been performed to measure corrosion of a nonreference, nuclear-grade graphite in deionized water and air (Refs. 1 and 2). The water tests were performed at 200 to 300 °C in a pressurized autoclave. Measurements of CO_2 and CO in water were used to quantify the corrosion rates. Both water and air reaction rates showed an Arrhenius temperature dependence. At 300 °C, the graphite oxidation rate was 100x higher in deionized water than in air, suggesting that the reaction rate in water may be controlled by a catalytic process involving dissolved oxygen. The leach rate of graphite in deionized water was more than a factor of 10^5 slower than the leach rate of waste glass. Tests were also performed on graphite while exposed to a Co-60 gamma source. The total dose over 24 hours ranged from 3.8×10^6 to 7.7×10^6 Roentgens. No significant radiolytic effects on the graphite corrosion rate were observed.

1.3 Data Needed

Data are needed for the corrosion rates of reference DB-MHR graphite during off-normal (groundwater ingress) conditions, including cycling between wet and dry conditions. In order to obtain these data and subsequent corrosion rate correlations, accelerated tests should be performed as a function of water chemistry, graphite temperature, and time. The variation in water chemistry

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* It is assumed that the reference fuel-element graphite for the DB-MHR will be the functional equivalent of H-451.

should cover the range from the distilled water to water similar in chemistry to Yucca Mountain Site groundwater. Some tests must be performed on irradiated graphite to investigate possible irradiation effects (the available data imply that the effect will be unimportant).

1.4 Parameter/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications as appropriate:

Test Articles

Graphite body	Reference fuel-element graphite; unit cell of prismatic element

Normal Repository Environment

Peak fast fluence	[$\leq 8 \times 10^{25}$] n/m^2 (E > 29 fJ)
Repository pressure	1 atm
Residence time	$\leq 10^6$ years
Fuel element temperatures	[20^o to 250 ^{o}C]
Environment (Air)	[$\leq$ 0.01 atm H_2O] [0.21 atm O_2] [0.79 atm N_2]

Off-Normal Conditions (Groundwater Ingress)

Groundwater chemistry	Yucca Mountain, [TBD}
Vapor-phase environment	$O_2/N_2/H_2O/CO/H_2$
Vapor-phase impurity levels	[0 to 1.0] atm H_2O
Fuel-element temperatures	[20 – 100 ^{o}C]
Groundwater contact time	TBD
Repository pressure	$\geq$ 1 atm
Expected failure fraction Off-normal conditions	[10^{-4} to 10^{-2}]

2. DESIGNER'S ALTERNATIVES

Alternatives to the acquisition of the above data are:

2.1 Use the existing database and add design margins to account for the uncertainties.

2.2 Use geologic analogies of long-term corrosion of graphite by ground water as it occurs in nature. Perform a literature search of data for naturally occurring groundwater corrosion of graphite and use such data to estimate the corrosion that can occur in fuel element graphite during the long-term storage under the groundwater ingress conditions.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Measure corrosion rates for the reference fuel-element graphite in groundwater in accelerated tests. The corrosion rates will be measured as a function of groundwater chemistry, fuel element graphite temperature, and time.

Alternative 2.1 is rejected because of large uncertainties associated with the extrapolation of the existing data base for significantly larger times and for different groundwater chemistries and temperatures. The current data base is from tests run for only 24 hours. The analysis of these test results will give average rates for that time period, but there is no way to differentiate initial rates from long-term rates; extrapolation in time over several orders of magnitude is highly uncertain.

Alternative 2.2 is not acceptable because the differences between naturally occurring graphite and nuclear grade graphite, especially regarding impurities, and between the conditions in nature and in a spent-fuel repository during off-normal conditions will result in large uncertainties in the estimates of groundwater corrosion of the H-451 graphite.

4. SCHEDULE REQUIREMENTS

TBD

5. PRIORITY

Urgency: TBD
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H

6. CONSEQUENCES OF NONEXECUTION AND FALLBACK POSITION

Accept Alternative 2.1 as the fallback position, which will result in large uncertainties in the calculation of the H-451 fuel element corrosion. This will, in turn, lead to large uncertainties in the calculation of corrosion and failure rates of the fuel compact matrix and the fuel particle coatings. A weakened position during safety review would result from these uncertainties.

7. REFERENCES

1. Gray, W. J., "Reaction of Graphite with Water and its Implications for Radioactive Waste Storage," Radioactive Waste Management, Vol. 1(1), May 1980, pp. 105-109.

2. Gray, W. J., "A study of the Oxidation of Graphite in Liquid Water for Radioactive Storage Applications," Radioactive Waste Management and the Nuclear Fuel Cycle, Vol. 3(2), December 1982, pp. 137-149.

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

“AAA” Fuel Program
C-14 CONTENT OF MATRIX AND GRAPHITE
DDN RTW.04.13

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

C-14 is a neutron activation product, which is produced in HTGR graphite structures by two reactions: (1) activation of N-14, via an (n,p) reaction, which exists as an impurity in graphite (N-14 is introduced during the manufacture of the fuel-element graphite and compact matrix), and (2) activation of C-13, which is a rare isotope that is naturally present in carbon-based structures. Because of its relatively high mobility, C-14 is one of the key radionuclides of concern for release by groundwater leaching. In addition, its relatively long half-life (5730 years) makes it a potential long-term health hazard. In order to calculate C-14 release rates, the inventories of C-14 in the spent fuel-element graphite and fuel compact matrix must be quantified.

1.1 Summary of Functions/Assumptions

[TBD]

1.2 Current Data Base Summary

The current data base for C-14 inventories in nuclear fuel-element graphite is limited and variable. The most extensive data set is probably that obtained for Peach Bottom 1 for which an attempt was made to estimate an overall C-14 mass balance for the reactor core (Ref. 1); the apparent range of N-14 concentrations in PB graphite (as inferred from the measured N-14 concentration and the calculated thermal neutron fluence) varied over two orders of magnitude, but an initial N-14 concentration well in excess of 100 ppm was inferred at certain core locations. More limited measurements were made of C-14 inventories in Fort St. Vrain (FSV) fuel elements and fuel test elements (Ref. 2). Based upon the measured C-14 contents and relatively crude estimates of the time-integrated neutron fluxes at the sample locations, the inferred initial N-14 content ranged from 10-30 ppm.

Experiments were also performed to measure C-14 leach rates from various irradiated graphites, including specimens from the German AVR (Ref. 3), French Magnox (Ref. 4), Hanford N-Reactor (Ref. 4), and British Magnox (Ref. 5) reactors, but the initial inventories of C-14 were not reported in all cases. Except for the data for the FSV fuel test element FTE-2, the C-14 data are for graphites other than H-451 graphite.[*] There are also some limited data on N-14 content in FSV fuel elements fabricated from H-327 graphite.

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[*] It is assumed that the reference fuel-element graphite for the DB-MHR will be the functional equivalent of H-451.

1.3 Data Needed

Data are needed to permit reliable quantification of the C-14 concentrations in irradiated, reference DB-MHR fuel-element graphite and in fuel-compact matrix. Tests should be performed to measure the C-14 inventories in the H-451 graphite irradiated in the FSV reactor and/or HTGR fuel irradiation capsules since it is anticipated that such H-451 data will provide a reliable basis for estimating the C-14 in reference DB-MHR graphite (assuming appropriate adjustments for differences in thermal fluences, etc.). Data are also needed for the N-14 content in as-manufactured graphite from archival specimens; this will allow the calculation of the C-14 inventory by neutron activation of N-14.

1.4 Parameter/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications as appropriate:

Test Articles

Matrix type	[Thermosetting resin]
Neutron poisons	PyC-coated B_4C PyC-coated Er_2O_3
Graphite body	Reference fuel-element graphite; H-451

Normal Repository Environment

Peak fast fluence	[$\leq 8 \times 10^{25}$] n/m^2 (E > 29 fJ)
Repository pressure	1 atm
Residence time	$\leq 10^6$ years
Fuel element temperatures	[20 to 250 ^{o}C]
Environment (Air)	[$\leq$ 0.01 atm H_2O] [0.21 atm O_2] [0.79 atm N_2]

Off-Normal Conditions (Groundwater Ingress)

Groundwater chemistry	Yucca Mountain, [TBD}
Vapor-phase environment	$O_2/N_2/H_2O/CO/H_2$
Vapor-phase impurity levels	[0 to 1.0] atm H_2O

Fuel-element temperatures	[20 – 100 ^{o}C]
Groundwater contact time	TBD
Repository pressure	≥1 atm

2. DESIGNER'S ALTERNATIVES

Alternatives to the acquisition of the above data are:

2.1 Use the existing data base and add design margin to account for the uncertainties.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Measure the N-14 and C-14 contents in a statistically significant number of samples of unirradiated and irradiated H-451 graphite and reference DB-MHR graphite and matrix (when available). As an early indicator, the C-14 contents should be measured in the H-451 graphite irradiated in the FSV reactor and/or HTGR fuel irradiation capsules.

Alternative 2.1 is rejected because of large uncertainties associated with the existing data base and because of the dominant contribution of C-14 to predicted environmental dose rates for a repository containing spent HTGR fuel elements. The existing data base is rather meager and it is primarily for graphites other than H-451.

4. SCHEDULE REQUIREMENTS

TBD

5. PRIORITY

Urgency: TBD
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H

6. CONSEQUENCES OF NONEXECUTION AND FALLBACK POSITION

Accept Alternative 2.1 as the fallback position, which will result in large uncertainties in the calculation of the C-14 inventories in DB-MHR fuel compact matrix and fuel element graphite. This will, in turn, lead to excessive conservatism in the calculation of C-14 release and the attendant predicted environmental dose rates for the repository.

7. REFERENCES

1. Wichner, R. P., and F. F. Dyer, "Carbon-14 Production in the Peach Bottom Core," ORNL-5597, Oak Ridge National Laboratory, July 1980.

2. Montgomery, F. C., and R. D. Burnette, “FSV Cycle 4 Coolant Chemistry and Tritium Sorption on Moderator Graphite,” HTGR-86-079, Rev. 0, GA Technologies, September 1986.

3. Zhang, Z., "Untersuchengen zum Nachweis der Langseitzicherheit bei der Endlagerung abgebrannter Hochtemperaturreaktor-Brennelemente in Untertagigen Gebirgsformationen (Investigations to Prove the Long Term Safety Aspects of the Final Disposition of HTR Fuel Elements in Underground Mountain Formations)," KFA Report JUL-2796 7/93, ISSN 0944-2952, Kernforschungsanlage, Jülich, Germany, 1993.

4. Gray, W. J., and W. C. Morgan, "Leaching of C-14 and Cl-36 from Irradiated French Graphite," PNL-6989, Battelle Pacific Northwest Laboratories, December 1989.

5. White, I. F., et al., "Assessment of Management Modes for Graphite from Reactor Decommissioning," EUR 9232, Commission of the European Communities, Contract No. DE-D-001-UK, 1984.

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

“AAA” Fuel Program
C-14 LEACHING RATES FROM MATRIX AND GRAPHITE
DDN RTW.04.14

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

C-14 is one of the key radionuclides of concern for release by groundwater leaching. C-14 is a neutron activation product, which is produced in HTGR graphite structures by two reactions: (1) activation of N-14, via an (n,p) reaction, which exists as an impurity in graphite (N-14 is introduced during the manufacture of the fuel-element graphite and compact matrix), and (2) activation of C-13, which is a rare isotope that is naturally present in carbon-based structures. During the postulated off-normal repository condition of groundwater ingress, C-14 can be released to groundwater by leaching of the fuel element graphite and compact matrix. Therefore, the C-14 leaching rates from the irradiated, reference DB-MHR fuel-element graphite* and compact matrix must be quantified.

1.1 Summary of Functions/Assumptions

[TBD]

1.2 Current Data Base Summary

Data on the leaching rates of C-14 from graphite irradiated in gas-cooled reactors is available from four tests. The observed leaching rates varied over several orders of magnitude. The highest leaching rates were observed for the French CO_2-cooled Magnox graphite; the values ranged from 0.004/y to 0.03/y (Ref. 1). This graphite probably experienced oxidation from the CO_2 coolant. Intermediate values, $\sim 5\times10^{-4}$/y, came from Hanford and British Magnox graphites (Refs 1 and 2, respectively). The British samples were also irradiated in a CO_2-cooled Magnox reactor, but any effects of oxidation were removed by machining of the surface layer prior to leaching. The lowest leaching rate, $\sim 1\times10^{-5}$/y, was for German fuel pebbles, irradiated in the He-cooled AVR (Ref. 3).

1.3 Data Needed

The current data base contains leaching rates from different graphites, but not from the H-451 graphite (which is considered a reliable surrogate for reference DB-MHR graphite). Data are needed for leaching rates of C-14 from the H-451 irradiated graphite and from the reference DB-MHR graphite and compact matrix (when available). Accelerated lab tests should be performed to measure C-14

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* It is assumed that the reference fuel-element graphite for the DB-MHR will be the functional equivalent of H-451.

leaching rates in the H-451 graphite irradiated in the Fort Saint Vrain reactor and/or HTGR test capsules. Tests should be performed as a function of water chemistry, temperature, and time.

1.4 Parameter/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications as appropriate:

Test Articles

Matrix type	[Thermosetting resin]
Neutron poisons	PyC-coated B_4C PyC-coated Er_2O_3
Graphite body	Reference fuel-element graphite; H-451

Normal Repository Environment

Peak fast fluence	[$\leq 8 \times 10^{25}$] n/m^2 (E > 29 fJ)
Repository pressure	1 atm
Residence time	$\leq 10^6$ years
Fuel element temperatures	[20 to 250 ^{o}C]
Environment (Air)	[$\leq$ 0.01 atm H_2O] [0.21 atm O_2] [0.79 atm N_2]

Off-Normal Conditions (Groundwater Ingress)

Groundwater chemistry	Yucca Mountain, [TBD}
Vapor-phase environment	$O_2/N_2/H_2O/CO/H_2$
Vapor-phase impurity levels	[0 to 1.0] atm H_2O
Fuel-element temperatures	[20 – 100 ^{o}C]
Groundwater contact time	TBD
Repository pressure	$\geq$ 1 atm

2. DESIGNER'S ALTERNATIVES

Alternatives to the acquisition of the above data are:

2.1 Use the existing data base and add design margins to account for the considerable uncertainties.

2.2 Assume the complete leaching of C-14 upon contact with groundwater.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Measure C-14 leaching rates of the H-451 irradiated graphite as an early indicator and from irradiated reference graphite and matrix when available. The C-14 contents should be measured in the H-451 graphite irradiated in the FSV reactor and the HTGR fuel test capsules.

Alternative 2.1 is rejected because of large uncertainties associated with the existing data base. Not only is there a large uncertainty in the current data base (the observed leaching rates vary over several orders of magnitude), but there are no data for the leaching rates for the irradiated H-451 graphite. Alternative 2.2 is rejected as excessively conservative, considering the important contribution of C-14 to the predicted environmental dose rates from the repository.

4. SCHEDULE REQUIREMENTS

TBD

5. PRIORITY

Urgency: TBD
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H

6. CONSEQUENCES OF NONEXECUTION AND FALLBACK POSITION

The fallback position would be Alternative 2.1, which will result in large uncertainties in the calculation of the C-14 leaching rates and release rates from DB-MHR fuel elements for the off-normal repository conditions. A weakened position during safety review would result from this uncertainty, and excessively conservative off-site dose rates would be predicted.

7. REFERENCES

1. Gray, W. J., and W. C. Morgan, "Leaching of C-14 and Cl-36 from Irradiated French Graphite," PNL-6989, Battelle Pacific Northwest Laboratories, December 1989.

2. White, I. F., et al., "Assessment of Management Modes for Graphite from Reactor Decommissioning," EUR 9232, Commission of the European Communities, Contract No. DE-D-001-UK, 1984.

3. Zhang, Z., "Untersuchengen zum Nachweis der Langseitzicherheit bei der Endlagerung abgebrannter Hochtemperaturreaktor-Brennelemente in Untertagigen Gebirgsformationen (Investigations to Prove the Long Term Safety Aspects of the Final Disposition of HTR Fuel Elements in Underground Mountain Formations)," KFA Report JUL-2796 7/93, ISSN 0944-2952, Kernforschungsanlage, Jülich, Germany, 1993.

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

"AAA" Fuel Program
CHEMICAL IMPURITIES IN MATRIX AND GRAPHITE
DDN RTW.04.15

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

Nuclear-grade graphites are by necessity high-purity materials because many of the chemical impurities that are typically present in the feedstocks of synthetic graphites are neutron poisons, and some of them catalyze graphite corrosion reactions, thereby limiting allowable service conditions. Fuel-compact matrix materials derived from thermosetting resins, which are expected to be used in the DB-MHR, are also typically high purity materials. Nevertheless, the fuel and graphite product specifications do allow low levels of impurities, and some of these residual impurities may be transmuted in the reactor to very long-lived radionuclides which can contribute significantly to predicted repository dose rates at long times. For example, in a British study of permanent disposal options for irradiated Magnox and AGR graphite core components, the dominant radionuclide for certain long-time scenarios was 300,000-yr Cl-36 which is produced by neutron activation of trace Cl impurities in the graphite. Consequently, the trace chemical impurities permitted in DB-MHR core graphite* and matrix materials must be thoroughly characterized.

The production of C-14, primarily via an (n,p) reaction of N-14 sorbed on core materials, is another example of an activated impurity. Because of the relatively high mobility of C-14 in the environment and its importance in determining environmental doses resulting from a geologic repository containing spent DB-MHR fuel elements, its production and transport properties are addressed in separate DDNs (DDN RTW.04.13 and DDN RTW.04.14, respectively).

1.1 Summary of Functions/Assumptions

[TBD]

1.2 Current Data Base Summary

The chemical impurities in various nuclear graphites, including the H-451 graphite used in two FSV fuel reload segments, have been controlled and characterized since the dawn of the nuclear age. As intimated above, the primary motivation has been twofold: (1) to limit the neutron poisons (several early graphite piles failed to go critical because of impurities in the graphite) and (2) to limit corrosion rates (the enhanced reactivity of PGX graphite used in the FSV core support structure due to high iron impurities became a major licensing issue).

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* It is assumed that the reference fuel-element graphite for the DB-MHR will be the functional equivalent of H-451.

Consequently, the fuel product specifications and graphite material specifications for various HTGR applications limit the chemical impurities in core materials in several ways: (1) explicit limits on certain neutron poisons, especially boron, expressed in terms of total allowed boron equivalents; (2) explicit limits on certain chemical impurities, such as iron, and classes of chemical impurities, such as transition metals; and (3) nonspecific limits on "total ash." In response, nuclear graphites, including H-451, and compact matrix materials have been characterized by various analytical methods to demonstrate compliance with these specifications (e.g., Ref. 1). However, these determinations typically have not achieved a complete mass balance, and the "ash" contents have not been fully characterized. Furthermore, the nuclear transmutation products resulting from exposing H-451 graphite and resin-derived fuel-compact matrix material to high neutron fluences have not been well characterized.

1.3 Data Needed

The chemical impurities in reference DB-MHR graphite and in resin-derived fuel-compact matrix material must be fully characterized. The chemical impurities in archival H-451 should be determined as an early indicator. The spectrum of nuclear transmutation products that can be generated from these impurities upon irradiation in the DB-MHR neutron spectrum must be determined by both direct measurement and analysis with appropriate nuclear depletion codes, such as GARGOYLE or ORIGIN. The test program shall include high-sensitivity gamma spectroscopy and radiochemistry of irradiated H-451 graphite specimens recovered from irradiated FSV fuel elements

1.4 Parameter/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications as appropriate:

Test Articles

Matrix type	[Thermosetting resin]
Neutron poisons	PyC-coated B_4C PyC-coated Er_2O_3
Graphite body	Reference fuel-element graphite; H-451

Normal Repository Environment

Peak fast fluence	[≤ 8 x 10^{25}] n/m^2 (E > 29 fJ)
Repository pressure	1 atm
Residence time	≤ 10^6 years

Fuel element temperatures	[20 to 250 °C]
Environment	
(Air)	[≤ 0.01 atm H_2O] [0.21 atm O_2] [0.79 atm N_2]

Off-Normal Conditions (Groundwater Ingress)

Groundwater chemistry	Yucca Mountain, [TBD}
Vapor-phase environment	$O_2/N_2/H_2O/CO/H_2$
Vapor-phase impurity levels	[0 to 1.0] atm H_2O
Fuel-element temperatures	[20 – 100 °C]
Groundwater contact time	TBD
Repository pressure	≥ 1 atm

2. DESIGNER'S ALTERNATIVES

Alternatives to the acquisition of the above data are:

2.1 Use the existing data base, especially the QC measurements for the H-451 graphite used for FSV reloads and the German data for their resin matrixes, and add design margins to account for the uncertainties.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Measure the chemical impurities in reference matrix and H-451 graphite.

Alternative 2.1 is rejected because of the large attendant uncertainties. The “total ash” component of H-451 graphite has not been fully characterized. The anticipated use of a new coke source for the future manufacture of H-451 equivalent nuclear graphite to be used in the DB-MHR may well introduce a different spectrum of chemical impurities into the final product. Finally, there is no guaranty that the chemical impurities in US-made, resin-derived, fuel-compact matrix would be identical to those in German-made resin matrix.

4. SCHEDULE REQUIREMENTS

TBD

5. PRIORITY

Urgency: TBD
Cost benefit: M
Uncertainty in existing data: M
Importance of new data: M

6. CONSEQUENCES OF NONEXECUTION AND FALLBACK POSITION

The fallback position would be to use Alternative 2.1, which will lead to large uncertainties in the calculation of the long-lived, nuclear transmutation products in core materials. While the total quantities of such transmutation products in spent fuel elements is small compared to the total radionuclide inventories contained in the fuel particles, the former can still contribute significantly to predicted repository source terms since they are not contained by intact particle coatings and certain of these products (e.g., Cl-36) are readily leached and transported by groundwater and have very long half-lifes. If these chemical impurities are not accurately and completely characterized, regulatory reviewers will likely mandate very conservative (and unfavorable) assumptions.

7. REFERENCES

1. Hoffman, B. L., and J. C. Scarborough, "Quality Assurance Aspects of the Fort St. Vrain Fuel", Proceedings of the International Conference on Nuclear Fuel Performance, October 15-19, 1973, London, England, British Nuclear Energy Society (CONF-731004), pp. 15A.1- 15A.6.

Originator Date

Engineering Manager Date

Project Manager Date

DATE: 09/27/02

“AAA” Fuel Program
RADIONUCLIDE LEACHING RATES FROM MATRIX AND GRAPHITE
DDN RTW.04.16

PLANT: DB-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The radionuclides in a spent HTGR fuel element are effectively retained (>99%) by the TRISO-coated fuel particles. Nevertheless, a small but important fraction of the radionuclides are found outside of the coated particles (as a result of contamination, failed particles, neutron activation, etc.) but sorbed on the fuel-compact matrix and graphite; these radionuclides are much more readily releasable from the fuel elements than those contained by intact TRISO coatings. In particular, during the postulated off-normal repository condition of groundwater ingress, these inventories can be released to groundwater by leaching of the graphite and matrix; however, the effective leaching rates may be limited by solubility limits and mass-transfer rates. The fission product leaching rates from irradiated DB-MHR core graphite[*] and matrix materials must be quantified.

1.1 Summary of Functions/Assumptions

[TBD]

1.2 Current Data Base Summary

Several tests have been performed to estimate the radionuclide leaching rates from nuclear-grade graphite. Graphite leaching tests were performed as part of decommissioning studies for the British graphite-moderated Magnox and Advanced Gas Reactors (Ref. 1). The tests were performed on small graphite samples that were irradiated in a CO_2-cooled Magnox reactor for 13 years. Prior to leaching, the samples were machined to remove the surface layer, so that the remaining radioactivity in the samples resulted almost entirely from nuclear reactions with impurities. Pre-test activity measurements showed significant quantities of the following nuclides: H-3, C-14, Fe-55, Co-60, Ba-133, Cs-134, Eu-154, and Eu-154. Leaching tests were performed using both groundwater and seawater. The groundwater tests were performed at 25 ^{o}C and a pressure of 1 bar for an exposure time of 150 days. The following nuclides were detected in the leachate samples: H-3, C-14, Co-60, Ba-133, and Cs-134. Activities of other nuclides were too low to be detected.

Tests were also performed to measure leaching rates of C-14 from graphite irradiated in the French Magnox reactor and from German pebble fuel elements

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[*] It is assumed that the reference fuel-element graphite for the DB-MHR will be the functional equivalent of H-451.

irradiated in the AVR. These test data are described in DDN RTW.04.14 dealing with C-14 leaching rates from graphite.

1.3 Data Needed

Data are needed for radionuclide leaching rates from irradiated, reference graphite and matrix. The leach rates from irradiated H-451 should be determined as an early indicator. Radionuclides of interest include fission products, activation products, and actinides, especially Pu isotopes. Accelerated lab tests should performed to measure leaching rates from the H-451 graphite irradiated in the Fort Saint Vrain (FSV) reactor, and/or HTGR fuel irradiation capsules. Tests should be performed as a function of water chemistry, temperature, and time.

1.4 Parameter/Service Conditions

Key test conditions are summarized below; test requirements will be elaborated in test specifications as appropriate:

Test Articles

Matrix type	[Thermosetting resin]
Neutron poisons	PyC-coated B_4C PyC-coated Er_2O_3
Graphite body	Reference fuel-element graphite; H-451

Normal Repository Environment

Peak fast fluence	[$\leq$ 8 x 10^{25}] n/m^2 (E > 29 fJ)
Repository pressure	1 atm
Residence time	$\leq 10^6$ years
Fuel element temperatures	[20 to 250 ^{o}C]
Environment	
(Air)	[$\leq$ 0.01 atm H_2O] [0.21 atm O_2] [0.79 atm N_2]

Off-Normal Conditions (Groundwater Ingress)

Groundwater chemistry	Yucca Mountain, [TBD}
Vapor-phase environment	$O_2/N_2/H_2O/CO/H_2$

Vapor-phase impurity levels	[0 to 1.0] atm H_2O
Fuel-element temperatures	[20 – 100 ^{o}C]
Groundwater contact time	TBD
Repository pressure	≥ 1 atm

2. DESIGNER'S ALTERNATIVES

Alternatives to the acquisition of the above data are:

2.1 Use the existing data base and add design margin to account for the uncertainties.

2.2 Assume complete leaching of all radionuclides sorbed on the fuel-compact matrix and graphite upon contact with groundwater and rely upon geological factors (low groundwater velocities, sorption, etc.) to retard the radionuclides sufficiently to show compliance with off-site dose limits.

3. SELECTED DESIGN APPROACH AND EXPLANATION

Measure radionuclide leaching rates from irradiated H-451 graphite and from reference graphite and compact matrix when available. As an early indicator, radionuclide leaching rates should be measured H-451 graphite specimens recovered from FSV fuel elements, and/or HTGR fuel-irradiation capsules.

Alternative 2.1 is rejected because of large uncertainties associated with extrapolating the current data base for the leaching rates from the British Magnox irradiated graphite to the irradiated H-451 graphite. The leach rate dependence on material properties and irradiation history has not been determined making the above extrapolation highly uncertain. Alternative 2.2 is rejected as being excessively conservative, but it may be feasible if more stringent limits were placed on the inventories of radionuclides occurring outside of intact TRISO-coated particles.

4. SCHEDULE REQUIREMENTS

TBD

5. PRIORITY

Urgency: TBD
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H

6. CONSEQUENCES OF NONEXECUTION AND FALLBACK POSITION

The fallback position would be to use Alternative 2.1, which will lead to large uncertainties in the calculation of radionuclide leaching rates and release rates from DB-MHR matrix and graphite for the off-normal repository conditions. A weakened position during safety review would result from this uncertainty; ultimately, reviewers may assume that all of the radionuclides sorbed on the matrix and graphite is leachable by groundwater. Depending upon other assumptions made by the reviewers, especially the degree of incremental coating failure in the repository at long times, compliance with off-site dose limits would eventually become problematic.

7. REFERENCES

1. White, I. F., et al., "Assessment of Management Modes for Graphite from Reactor Decommissioning," EUR 9232, Commission of the European Communities, Contract No. DE-D-001-UK, 1984.

Originator Date

Engineering Manager Date

Project Manager Date

Appendix B: Fuel Separations Interface Requirements

These interface requirements refer to the reprocessing of spent Driver Fuel and separation of the TRU material for use in fabrication of Transmutation Fuel. Separations efficiency goals are important parameters that must be accomplished during reprocessing of TRISO-coated fuel. This Reprocessing Interface Specification describes these goals in the context of the available methods for achieving them.

Initial head-end reprocessing unit operations must ensure access to the heavy metal oxides in the fuel kernels to allow their dissolution to form nitrates in strong acid solutions. These nitrate solutions may then be treated via conventional solvent extraction methods that have long been used for separating heavy metals and fission products.

The separated streams may then be further processed via denitration to yield feedstock for fabrication of Transmutation fuel.

Note that the following processing goals apply to the overall head-end reprocessing flowsheet, thus allowing ample opportunity for process development efforts to identify those combinations of unit operations whereby separations are best achieved. An example is in the removal of insoluble particulates from the dissolver product stream. While the overall specification may be stringent, the best process solution may include an initial coarse filter that is tolerant of high solids loadings, followed by a polishing filter that meets the product specification only if its inlet loading is relatively low (i.e. a cartridge membrane filter).

Conceptual Process Flowsheet

For reference, the following flowsheet is currently envisioned as a potential candidate for head-end reprocessing:

1. Remove fuel compacts
   a. Machine and/or saw ends and push out compacts (back-up is annular hole saw)
   b. GA's experience has been verified with irradiated fuel elements from FSV
2. Crush compacts/particles to < 30 microns
   a. Rough crush (jaw and/or roll crusher) followed by a steam fluid-energy mill to achieve <30 micron product
3. Acid leach
   a. Steam-sparged vertical batch dissolver with automated materials handling
   b. Operation draws upon GA's full-scale demonstration with very difficult fuel kernels (ThO_2)
4. Liquid-solid separation
   a. Acid-resistant continuous vacuum-belt filter coupled with a small polishing filter

Head-End Reprocessing Feed Specifications

- Prismatic fuel elements (dimensions – 800 mm in length, 360 mm across hexagonal cross-section flats)

- Fuel compacts (dimensions – 12.3 mm in diameter and 50 mm long)
- Driver Fuel particles are described in Table 4-3, and Transmutation Fuel particles are described in Table 4-4.
- Elemental composition per Deep Burn Transmutation Mass Flow charts (see Figure 2-2 for typical heavy nuclide composition)
- <1% of fuel compacts are broken prior to push-out
- 1-2 % of TRISO fuel particles are broken prior to push-out
- Provision for removing stacks of compacts which cannot be pushed out of the blocks will be provided

Head-End Reprocessing Product Specifications

- Nitrate solution containing >99.8% of all feed heavy metals (note: this % recovery goal is a stand-in pending clear guidance from overall economic studies); >95% of all feed fission products
- Maximum suspended solids content 0.1 mg/L
- Solvent extraction and denitration are assumed to be done in conventional processing, which would include recovery/reuse of nitric acid and solvents

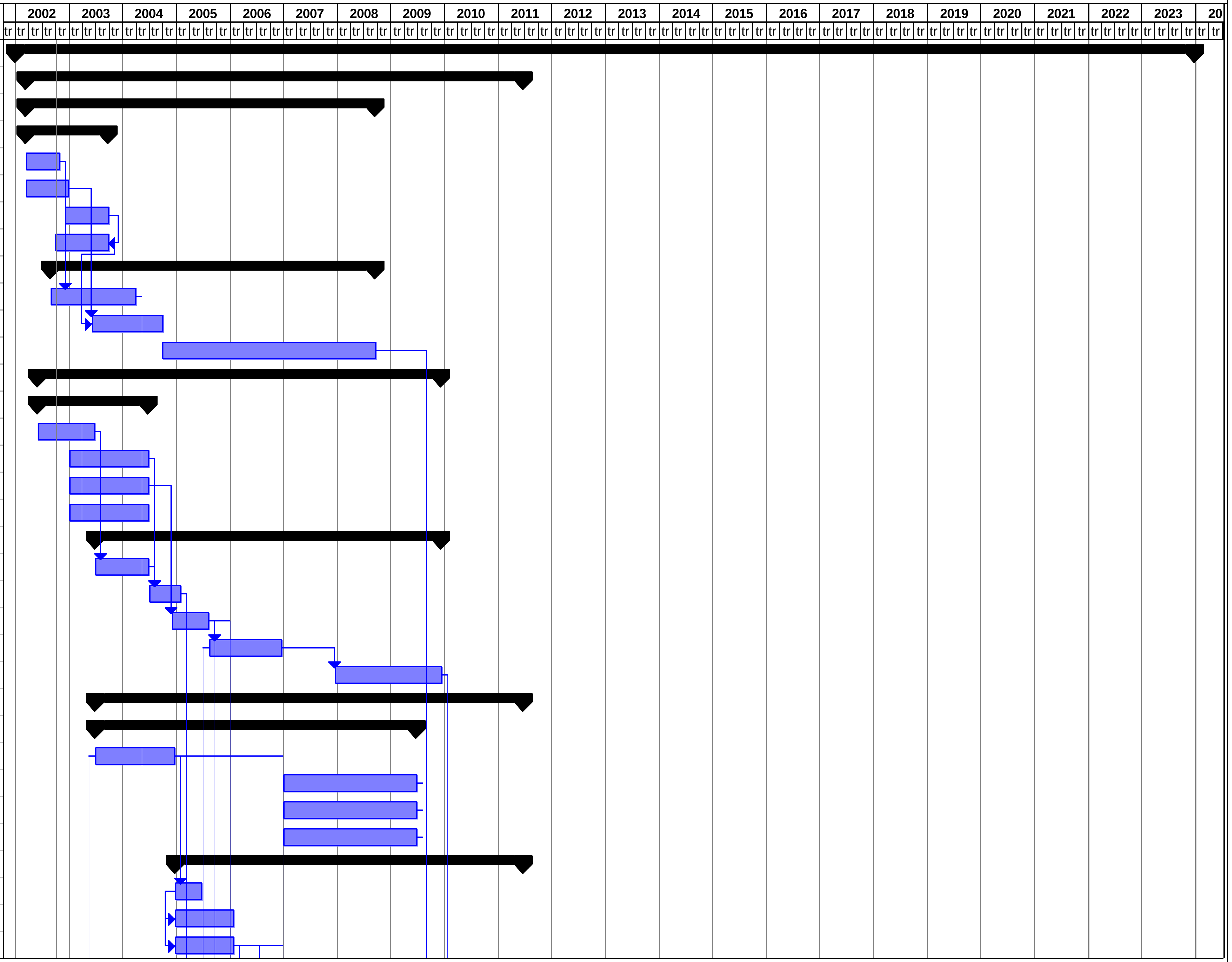
Head-End Reprocessing Waste Disposal Specifications

- Moisture content of solid residues <1%
- Goal (not requirement) is to reform compacts with the waste solids (C + SiC) with a minimal quantity of binder
- Fuel element may have reuse potential for waste disposal; therefore, it should be left intact to the extent possible (while allowing fuel compact push-out)
- All carbon to leave as a solid; i.e., no carbon dioxide (from combustion)
- Any off-gases to must be treated to meet nuclear regulatory and environmental constraints

Appendix C: Detailed Development Schedule

Development Schedule - Coated Particles for Transmutation

ID	Task Name	Start	Finish
1	**Fuel Development Schedule**	**1/1/02**	**12/29/23**
2	**Fuel Fabrication Process Development**	**3/14/02**	**6/30/11**
3	**Base Fuel Fabrication Technology**	**3/14/02**	**9/25/08**
4	**Uranium Fabrication Lab**	**3/14/02**	**9/30/03**
5	Install Uranium Kernel Line	3/14/02	10/31/02
6	Design/Install Sintering and Coating Line	3/14/02	12/31/02
7	Design/Install Compacting Line	12/2/02	9/30/03
8	Install Complete QC	10/1/02	9/30/03
9	**U-Test Fuel Fabrication**	**9/2/02**	**9/25/08**
10	Depleted Uranium Operations	9/2/02	3/31/04
11	Enriched Uranium Operation	6/3/03	10/1/04
12	Remote Process Development Support	10/1/04	9/25/08
13	**Driver Fuel (DF)**	**6/3/02**	**12/15/09**
14	**DF Fabrication Lab (glovebox)**	**6/3/02**	**6/30/04**
15	Design/Install DF Kernel Line	6/3/02	6/30/03
16	Design/Install DF Coater Line	1/1/03	6/30/04
17	Design/Install DF Compact Equipment	1/1/03	6/30/04
18	Establish Complete QC for DF	1/1/03	6/30/04
19	**DF Fabrication**	**7/1/03**	**12/15/09**
20	Produce DF Kernels	7/1/03	6/30/04
21	Produce DF TRISO Particles	7/1/04	2/1/05
22	Produce DF TRISO Compacts	12/1/04	8/16/05
23	Produce DF Margin Test Fuel	8/17/05	12/19/06
24	Produce DF Validation Fuel	12/19/07	12/15/09
25	**Transmutation Fuel (TF)**	**7/1/03**	**6/30/11**
26	**Remote Fuel Fabrication Line (hot cell)**	**7/1/03**	**7/2/09**
27	Design/Install TF Resin-loading Kernel Lin	7/1/03	12/27/04
28	Design/Install TF Coater Line	1/1/07	7/2/09
29	Design/Install TF Compact Equipment	1/1/07	7/2/09
30	Establish Complete QC for TF	1/1/07	7/2/09
31	**TF Fabrication**	**12/28/04**	**6/30/11**
32	Produce TF Pu/Am/Cm Kernels (hot cell)	12/28/04	6/27/05
33	Produce TF Pu/Am/Cm TRISO Particles (	12/28/04	1/25/06
34	Produce TF Pu/Am/Cm TRISO Compacts	12/28/04	1/25/06

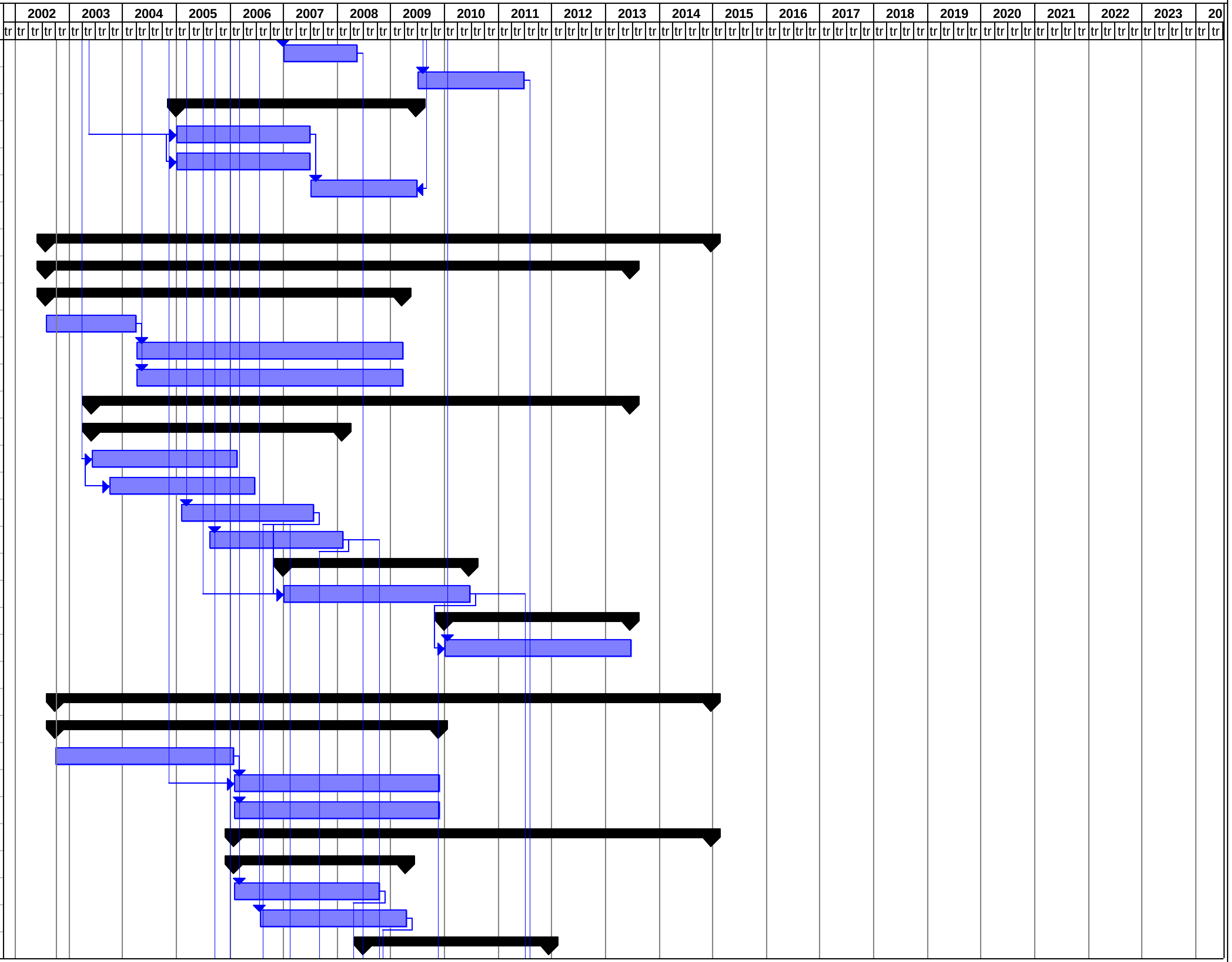


Project: AAA Development Plan- All Y
Date: 10/10/02

Task	▭	Milestone	◆	Rolled Up Task	▭	Rolled Up Progress	▬
Progress	▬	Summary	▬	Rolled Up Milestone	◇		

Development Schedule - Coated Particles for Transmutation

ID	Task Name	Start	Finish
35	Produce TF Margin Test Fuel (sparse)	12/28/06	5/14/08
36	Produce TF Validation Fuel (hot cell)	7/3/09	6/30/11
37	**Prototypic Scale Refabrication Analyses**	**1/3/05**	**6/26/09**
38	Preliminary Design of Industrial Scale Refabric	1/3/05	6/29/07
39	Equipment Analyses for Prototype Refabricati	1/3/05	6/29/07
40	Economics of Industrial Scale Refabrication O	7/2/07	6/26/09
41			
42	**Fuel Materials Development**	**8/1/02**	**12/29/14**
43	**Driver Fuel Evaluation and Testing**	**8/1/02**	**6/25/13**
44	**Non-Irradiation Evaluation of Candidates**	**8/1/02**	**3/25/09**
45	Preliminary DF Themochemical Analysis	8/1/02	3/31/04
46	DF Thermochemical Performance	4/1/04	3/25/09
47	DF Fuel Compact Thermophysical Proper	4/1/04	3/25/09
48	**Driver Fuel Irradiation Tests**	**6/3/03**	**6/25/13**
49	**Phase I - Screening of DF Candidate**	**6/3/03**	**2/11/08**
50	Uranium TRISO Benchmark Test	6/3/03	2/23/06
51	Uranium Compact Benchmark Test	10/1/03	6/23/06
52	ZrC-SiC DF TRISO Test	2/2/05	7/30/07
53	ZrC-SiC DF Compact Test	8/17/05	2/11/08
54	**Phase II - Design Improvements and**	**1/2/07**	**6/24/10**
55	DF Margin Tests- Multi-Cell (1)	1/2/07	6/24/10
56	**Phase III - Fuel Validation Test**	**1/1/10**	**6/25/13**
57	DF Validation Multi-Cell Capsule (1)	1/1/10	6/25/13
58			
59	**Transmutation Fuel Evaluation and Testing**	**10/1/02**	**12/29/14**
60	**Non-Irradiation Evaluation of TF Candida**	**10/1/02**	**11/25/09**
61	Preliminary TF Themochemical Analysis	10/1/02	1/25/06
62	TF Thermochemical Performance	1/26/06	11/25/09
63	TF Fuel Compact Thermophysical Propert	1/26/06	11/25/09
64	**Transmutation Fuel Irradiation Tests**	**1/26/06**	**12/29/14**
65	**TF Phase I - Screening of TF Candid**	**1/26/06**	**4/20/09**
66	TF1 Compact- Primary Fuel Design	1/26/06	10/20/08
67	TF2 Compact- Backup Fuel Design	7/27/06	4/20/09
68	**TF Phase II - Design Improvements a**	**7/1/08**	**12/23/11**

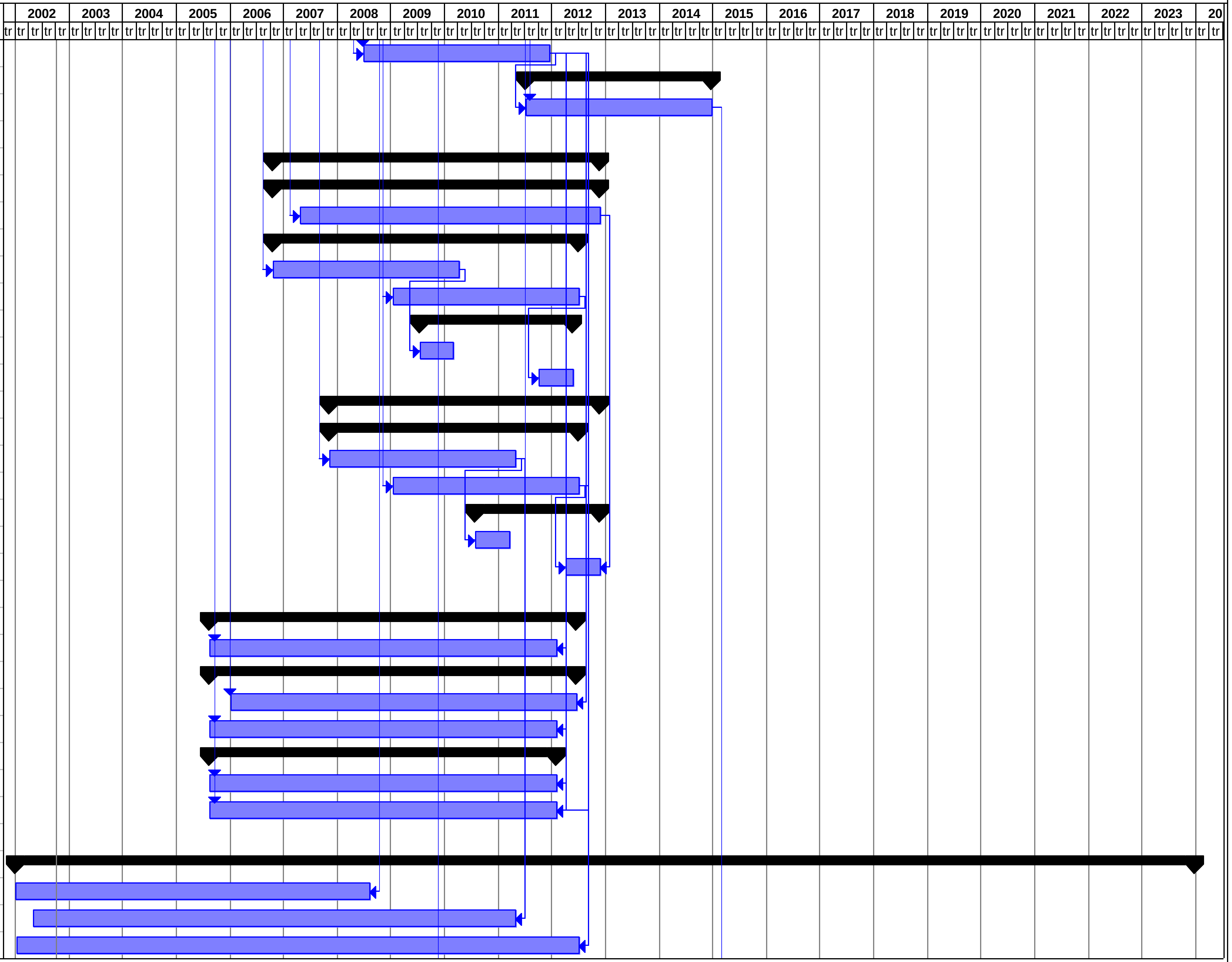


Project: AAA Development Plan- All Y
Date: 10/10/02

Task	▬	Milestone	◆	Rolled Up Task	▬	Rolled Up Progress	▬
Progress	▬	Summary	▬	Rolled Up Milestone	◇		

Development Schedule - Coated Particles for Transmutation

ID	Task Name	Start	Finish
69	TF Margin Tests- Multi-Cell (1)	7/1/08	12/23/11
70	**TF Phase III - Fuel Validation and Prc**	**7/6/11**	**12/29/14**
71	TF Validation Multi-Cell Capsule (1)	7/6/11	12/29/14
72			
73	**Fission Product and Actinide Transport**	**10/24/06**	**11/28/12**
74	**Retention in Fuel Particles**	**10/24/06**	**11/28/12**
75	Chemical Evaluation of Fission Products in DF	4/25/07	11/28/12
76	**Retention in DF and TF Particles**	**10/24/06**	**7/10/12**
77	FP-1 Retention in DF Particles- Normal Op	10/24/06	4/16/10
78	FP-2 Retention in TF Particles- Normal Op	1/15/09	7/10/12
79	**Accident Condition Testing of DF and TF**	**7/13/09**	**5/29/12**
80	Accident Condition Testing of DF Particle:	7/13/09	3/5/10
81	Accident Condition Testing of TF Particles	10/5/11	5/29/12
82	**Retention in Fuel Compacts**	**11/8/07**	**11/28/12**
83	**Retention in DF and TF Compacts**	**11/8/07**	**7/9/12**
84	FP-3 Retention in DF Compacts- Normal C	11/8/07	5/3/11
85	FP-4 Retention in TF Compacts- Normal C	1/14/09	7/9/12
86	**Accident Condition Testing of DF and TF**	**7/28/10**	**11/28/12**
87	Accident Condition Testing of DF Compac	7/28/10	3/22/11
88	Accident Condition Testing of TF Compac	4/5/12	11/28/12
89			
90	**Spent Fuel Disposal Technology**	**8/17/05**	**6/25/12**
91	Mechanical Integrity	8/17/05	2/8/12
92	**Long-Term Performance of Fuel Component**	**8/17/05**	**6/25/12**
93	Fuel Component Performance in Air	1/2/06	6/25/12
94	Fuel Component Performance in Groundwater	8/17/05	2/8/12
95	**Radionuclide Transport**	**8/17/05**	**2/8/12**
96	Radionuclide Inventories in Spent Fuel	8/17/05	2/8/12
97	Radionuclide Leaching Rates from Fuel Compo	8/17/05	2/8/12
98			
99	**Project Phases**	**1/1/02**	**12/29/23**
100	Fuel Demonstration	1/1/02	8/12/08
101	Proof of Principle- DF	5/2/02	5/3/11
102	Proof of Principle- TF	1/9/02	7/9/12

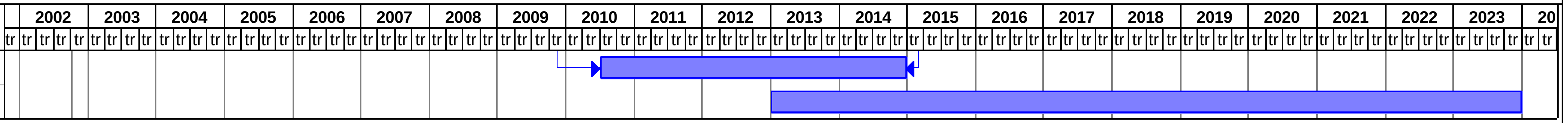


Project: AAA Development Plan- All Y
Date: 10/10/02

Task	🟦	Milestone	◆	Rolled Up Task	🟦	Rolled Up Progress	▬
Progress	▬	Summary	▬	Rolled Up Milestone	◇		

Development Schedule - Coated Particles for Transmutation

ID	Task Name	Start	Finish
103	Validation Tests of Fuel	7/2/10	12/29/14
104	Modular Reactor-DF/TF Demonstration Period	1/1/13	12/29/23



Project: AAA Development Plan- All Y
Date: 10/10/02

Task	▬	Milestone	◆	Rolled Up Task	▬	Rolled Up Progress	▬
Progress	▬	Summary	▬	Rolled Up Milestone	◇		

Appendix D: Detailed Cost Estimate

AAA Fuel Development Program Estimate- All Costs through Validation

WBS	Name	Duration	Start_Date	Finish_Date	Total All Costs	2002	2003	2004	2005	2006	2007	2008	2009	2010	2011	2012	2013	2014	2015
1	Fuel Development Schedule	5739 days	01/01/02	12/29/23	215,649,983	2,194,318	15,312,003	18,649,520	22,775,100	26,135,250	25,038,925	24,033,325	23,519,788	19,534,525	16,028,400	11,650,580	5,904,000	3,052,000	1,822,250
1.1	Fuel Fabrication Process Development	2426 days	03/14/02	06/30/11	75,257,675	2,092,293	11,958,358	11,443,200	11,210,775	9,440,800	7,926,500	6,636,250	6,410,750	5,168,250	2,970,500	-	-	-	-
1.1.1	Base Fuel Fabrication Technology	1706 days	03/14/02	09/25/08	17,879,525	1,803,543	3,786,983	2,835,000	3,672,500	2,877,500	1,936,000	968,000	-	-	-	-	-	-	-
1.1.1.1	Uranium Fabrication Lab	404 days	03/14/02	09/30/03	3,192,805	1,415,848	1,776,958	-	-	-	-	-	-	-	-	-	-	-	-
1.1.1.1.1	Install Uranium Kernel Line	166 days	03/14/02	10/31/02	295,170	295,170	-	-	-	-	-	-	-	-	-	-	-	-	-
1.1.1.1.2	Design/Install Sintering and Coating Line	209 days	03/14/02	12/31/02	746,720	746,720	-	-	-	-	-	-	-	-	-	-	-	-	-
1.1.1.1.3	Design/Install Compacting Line	217 days	12/02/02	09/30/03	792,915	61,958	730,958	-	-	-	-	-	-	-	-	-	-	-	-
1.1.1.1.4	Install Complete QC	261 days	10/01/02	09/30/03	1,358,000	372,000	1,046,000	-	-	-	-	-	-	-	-	-	-	-	-
1.1.1.2	U-Test Fuel Fabrication	1584 days	09/02/02	09/25/08	5,543,720	387,695	2,010,025	2,351,000	795,000	-	-	-	-	-	-	-	-	-	-
1.1.1.2.1	Depleted Uranium Operations	413 days	09/02/02	03/31/04	3,577,720	387,695	1,427,025	1,373,000	390,000	-	-	-	-	-	-	-	-	-	-
1.1.1.2.2	Enriched Uranium Operation	349 days	06/03/03	10/01/04	2,450,000	-	583,000	1,269,500	597,500	-	-	-	-	-	-	-	-	-	-
1.1.1.2.3	Remote Process Development Support	1040 days	10/01/04	09/25/08	9,143,000	-	-	484,000	2,877,500	2,877,500	1,936,000	968,000	-	-	-	-	-	-	-
1.1.2.	Driver Fuel (DF)	1967 days	06/03/02	12/15/09	23,576,675	288,750	6,826,750	6,512,200	2,135,475	3,126,500	-	145,750	2,343,500	2,197,750	-	-	-	-	-
1.1.2.1	DF Fabrication Lab (glovebox)	543 days	06/03/02	06/30/04	11,738,625	288,750	6,459,750	4,990,125	-	-	-	-	-	-	-	-	-	-	-
1.1.2.1.1	Design/Install DF Kernel Line	281 days	06/03/02	06/30/03	2,023,500	288,750	1,096,750	638,000	-	-	-	-	-	-	-	-	-	-	-
1.1.2.1.2	Design/Install DF Coater Line	391 days	01/01/03	06/30/04	5,329,000	-	2,679,500	2,649,500	-	-	-	-	-	-	-	-	-	-	-
1.1.2.1.3	Design/Install DF Compact Equipment	391 days	01/01/03	06/30/04	1,667,625	-	1,101,750	565,875	-	-	-	-	-	-	-	-	-	-	-
1.1.2.1.4	Establish Complete QC for DF	391 days	01/01/03	06/30/04	2,718,500	-	1,581,750	1,136,750	-	-	-	-	-	-	-	-	-	-	-
1.1.2.2	DF Fabrication	1686 days	07/01/03	12/15/09	11,838,050	-	367,000	1,522,075	2,135,475	3,126,500	-	145,750	2,343,500	2,197,750	-	-	-	-	-
1.1.2.2.1	Produce DF Kernels	262 days	07/01/03	06/30/04	734,000	-	367,000	367,000	-	-	-	-	-	-	-	-	-	-	-
1.1.2.2.2	Produce DF TRISO Particles	154 days	07/01/04	02/01/05	1,203,000	-	-	1,038,275	164,725	-	-	-	-	-	-	-	-	-	-
1.1.2.2.3	Produce DF TRISO Compacts	185 days	12/01/04	08/16/05	1,057,800	-	-	116,800	941,000	-	-	-	-	-	-	-	-	-	-
1.1.2.2.4	Produce DF Margin Test Fuel	350 days	08/17/05	12/19/06	4,156,250	-	-	-	1,029,750	3,126,500	-	-	-	-	-	-	-	-	-
1.1.2.2.5	Produce DF Validation Fuel	520 days	12/19/07	12/15/09	4,687,000	-	-	-	-	-	-	145,750	2,343,500	2,197,750	-	-	-	-	-
1.1.3	Transmutation Fuel (TF)	2088 days	07/01/03	06/30/11	26,510,475	-	1,344,625	2,096,000	2,752,800	786,800	4,874,000	4,939,500	3,775,750	2,970,500	2,970,500	-	-	-	-
1.1.3.1	Remote Fuel Fabrication Line (hot cell)	1568 days	07/01/03	07/02/09	13,725,125	-	1,344,625	1,976,000	-	-	2,803,000	3,825,750	3,775,750	-	-	-	-	-	-
1.1.3.1.1	Design/Install TF Resin-loading Kernel Line	390 days	07/01/03	12/27/04	3,320,625	-	1,344,625	1,976,000	-	-	-	-	-	-	-	-	-	-	-
1.1.3.1.2	Design/Install TF Coater Line	654 days	01/01/07	07/02/09	4,577,000	-	-	-	-	-	1,144,000	1,716,500	1,716,500	-	-	-	-	-	-
1.1.3.1.3	Design/Install TF Compact Equipment	654 days	01/01/07	07/02/09	2,489,750	-	-	-	-	-	626,250	956,750	906,750	-	-	-	-	-	-
1.1.3.1.4	Establish Complete QC for TF	654 days	01/01/07	07/02/09	3,337,750	-	-	-	-	-	1,032,750	1,152,500	1,152,500	-	-	-	-	-	-
1.1.3.2	TF Fabrication	1698 days	12/28/04	06/30/11	12,785,350	-	-	120,000	2,752,800	786,800	2,071,000	1,113,750	-	2,970,500	2,970,500	-	-	-	-
1.1.3.2.1	Produce TF Pu/Am/Cm Kernels (hot cell)	130 days	12/28/04	06/27/05	664,000	-	-	120,000	544,000	-	-	-	-	-	-	-	-	-	-
1.1.3.2.2	Produce TF Pu/Am/Cm TRISO Particles (sparse)	282 days	12/28/04	01/25/06	1,347,800	-	-	-	987,800	360,000	-	-	-	-	-	-	-	-	-
1.1.3.2.3	Produce TF Pu/Am/Cm TRISO Compacts (sparse)	282 days	12/28/04	01/25/06	1,347,800	-	-	-	1,221,000	126,800	-	-	-	-	-	-	-	-	-
1.1.3.2.4	Produce TF Margin Test Fuel (sparse)	360 days	12/28/06	05/14/08	3,484,750	-	-	-	-	300,000	2,071,000	1,113,750	-	-	-	-	-	-	-
1.1.3.2.5	Produce TF Validation Fuel (hot cell)	520 days	07/03/09	06/30/11	5,941,000	-	-	-	-	-	-	-	-	2,970,500	2,970,500	-	-	-	-
1.1.4	Prototypic Scale Refabrication Analyses	1170 days	01/03/05	06/26/09	7,291,000	-	-	-	2,650,000	2,650,000	1,116,500	583,000	291,500	-	-	-	-	-	-
1.1.4.1	Preliminary Design of Industrial Scale Refabrication Plant	650 days	01/03/05	06/29/07	4,667,500	-	-	-	2,067,000	2,067,000	533,500	-	-	-	-	-	-	-	-
1.1.4.2	Equipment Analyses for Prototype Refabrication Equipment	650 days	01/03/05	06/29/07	1,457,500	-	-	-	583,000	583,000	291,500	-	-	-	-	-	-	-	-
1.1.4.3	Economics of Industrial Scale Refabrication Operations	520 days	07/02/07	06/26/09	1,166,000	-	-	-	-	-	291,500	583,000	291,500	-	-	-	-	-	-
1.2	Fuel Materials Development	3369 days	08/01/02	06/30/15	100,708,278	102,025	3,353,645	7,206,320	11,564,325	15,225,250	13,944,975	12,339,475	8,891,788	5,552,225	5,104,000	6,646,000	5,904,000	3,052,000	1,822,250
1.2.1	Driver Fuel Evaluation and Testing	2976 days	08/01/02	12/26/13	59,065,728	43,725	3,062,145	6,914,820	11,272,825	10,454,400	7,946,125	5,022,625	2,941,063	2,752,000	2,452,000	3,352,000	2,852,000	-	-
1.2.1.1	Non-Irradiation Evaluation of Candidates	1735 days	08/01/02	03/25/09	3,993,658	43,725	145,750	938,620	931,625	581,625	581,625	581,625	189,063	-	-	-	-	-	-
1.2.1.1.1	Preliminary DF Themochemical Analysis	435 days	08/01/02	03/31/04	227,370	43,725	145,750	37,895	-	-	-	-	-	-	-	-	-	-	-
1.2.1.1.2	DF Thermochemical Performance	1300 days	04/01/04	03/25/09	2,393,563	-	-	570,625	562,750	387,750	387,750	387,750	96,938	-	-	-	-	-	-
1.2.1.1.3	DF Fuel Compact Thermophysical Properties	1300 days	04/01/04	03/25/09	1,372,725	-	-	330,100	368,875	193,875	193,875	193,875	92,125	-	-	-	-	-	-
1.2.1.2	Driver Fuel Irradiation Tests	2757 days	06/03/03	12/26/13	55,072,070	-	2,916,395	5,976,200	10,341,200	9,872,775	7,364,500	4,441,000	2,752,000	2,752,000	2,452,000	3,352,000	2,852,000	-	-
1.2.1.2.1	Phase I - Screening of DF Candidates	1356 days	06/03/03	08/12/08	36,299,570	-	2,916,395	5,976,200	10,341,200	9,872,775	5,204,000	1,989,000	-	-	-	-	-	-	-
1.2.1.2.1.1	Uranium TRISO Benchmark Test	713 days	06/03/03	08/28/06	13,718,570	-	2,353,395	4,024,200	5,024,200	2,316,775	-	-	-	-	-	-	-	-	-
1.2.1.2.1.2	Uranium Compact Benchmark Test	713 days	10/01/03	12/26/06	7,769,000	-	563,000	1,952,000	2,602,000	2,652,000	-	-	-	-	-	-	-	-	-
1.2.1.2.1.3	ZrC-SiC DF TRISO Test	649 days	02/02/05	01/29/08	7,406,000	-	-	-	2,152,000	2,602,000	2,652,000	-	-	-	-	-	-	-	-
1.2.1.2.1.4	ZrC-SiC DF Compact Test	649 days	08/17/05	08/12/08	7,406,000	-	-	-	563,000	2,302,000	2,552,000	1,989,000	-	-	-	-	-	-	-
1.2.1.2.2	Phase II - Design Improvements and Evaluation	1039 days	01/02/07	12/27/10	10,116,500	-	-	-	-	-	2,160,500	2,452,000	2,752,000	2,752,000	-	-	-	-	-
1.2.1.2.2.1	DF Margin Tests- Multi-Cell (1)	908 days	01/02/07	12/27/10	10,116,500	-	-	-	-	-	2,160,500	2,452,000	2,752,000	2,752,000	-	-	-	-	-
1.2.1.2.3	Phase III - Fuel Validation Test	1039 days	01/01/10	12/26/13	8,656,000	-	-	-	-	-	-	-	-	-	2,452,000	3,352,000	2,852,000	-	-
1.2.1.2.3.1	DF Validation Multi-Cell Capsule (1)	908 days	01/01/10	12/26/13	8,656,000	-	-	-	-	-	-	-	-	-	2,452,000	3,352,000	2,852,000	-	-
1.2.2	Transmutation Fuel Evaluation and Testing	3326 days	10/01/02	06/30/15	41,642,550	58,300	291,500	291,500	291,500	4,770,850	5,998,850	7,316,850	5,950,725	2,800,225	2,652,000	3,294,000	3,052,000	3,052,000	1,822,250
1.2.2.1	Non-Irradiation Evaluation of TF Candidates	1867 days	10/01/02	11/25/09	4,560,300	58,300	291,500	291,500	291,500	1,144,725	1,094,850	619,850	619,850	148,225	-	-	-	-	-
1.2.2.1.1	Preliminary TF Themochemical Analysis	867 days	10/01/02	01/25/06	1,078,550	58,300	291,500	291,500	291,500	145,750	-	-	-	-	-	-	-	-	-
1.2.2.1.2	TF Thermochemical Performance	1000 days	01/26/06	11/25/09	2,085,600	-	-	-	-	614,625	662,750	387,750	387,750	32,725	-	-	-	-	-

WBS	Name	Duration	Start_Date	Finish_Date	Total All Costs	2002	2003	2004	2005	2006	2007	2008	2009	2010	2011	2012	2013	2014	2015
1.2.2.1.3	TF Fuel Compact Thermophysical Properties	1000 days	01/26/06	11/25/09	1,396,150	-	-	-	-	384,350	432,100	232,100	232,100	115,500	-	-	-	-	-
1.2.2.2	Transmutation Fuel Irradiation Tests	2459 days	01/26/06	06/30/15	37,082,250	-	-	-	-	3,626,125	4,904,000	6,697,000	5,330,875	2,652,000	2,652,000	3,294,000	3,052,000	3,052,000	1,822,250
1.2.2.2.1	TF Phase I - Screening of TF Candidates	974 days	01/26/06	10/21/09	16,338,000	-	-	-	-	3,626,125	4,904,000	5,129,000	2,678,875	-	-	-	-	-	-
1.2.2.2.1.1	TF1 Compact- Primary Fuel Design	713 days	01/26/06	04/22/09	8,169,000	-	-	-	-	2,352,000	2,552,000	2,552,000	713,000	-	-	-	-	-	-
1.2.2.2.1.2	TF2 Compact- Backup Fuel Design	713 days	07/27/06	10/21/09	8,169,000	-	-	-	-	1,274,125	2,352,000	2,577,000	1,965,875	-	-	-	-	-	-
1.2.2.2.2	TF Phase II - Design Improvements and Evaluation	1040 days	07/01/08	06/25/12	11,050,000	-	-	-	-	-	-	1,568,000	2,652,000	2,652,000	2,652,000	1,526,000	-	-	-
1.2.2.2.2.1	TF Margin Tests- Multi-Cell (1)	909 days	07/01/08	06/25/12	11,050,000	-	-	-	-	-	-	1,568,000	2,652,000	2,652,000	2,652,000	1,526,000	-	-	-
1.2.2.2.3	TF Phase III - Fuel Validation and Proof Tests	1040 days	07/06/11	06/30/15	9,694,250	-	-	-	-	-	-	-	-	-	-	1,768,000	3,052,000	3,052,000	1,822,250
1.2.2.2.3.1	TF Validation Multi-Cell Capsule (1)	909 days	07/06/11	06/30/15	9,694,250	-	-	-	-	-	-	-	-	-	-	1,768,000	3,052,000	3,052,000	1,822,250
1.3	Fission Product and Actinide Transport	1491 days	04/24/07	01/09/13	32,876,650	-	-	-	-	-	2,198,250	4,088,400	7,248,050	7,844,850	6,984,700	4,512,400	-	-	-
1.3.1	Retention in Fuel Particles	1491 days	04/24/07	01/09/13	17,850,075	-	-	-	-	-	1,733,800	2,262,825	3,921,800	4,439,450	3,348,550	2,143,650	-	-	-
1.3.1.1	Chemical Evaluation of Fission Products in DF and TF	1461 days	04/25/07	11/28/12	2,623,500	-	-	-	-	-	349,800	524,700	524,700	524,700	524,700	174,900	-	-	-
1.3.1.2	Retention in DF and TF Particles	1491 days	04/24/07	01/09/13	13,835,325	-	-	-	-	-	1,384,000	1,738,125	3,397,100	3,376,250	2,266,725	1,673,125	-	-	-
1.3.1.2.1	FP-1 Retention in DF Particles- Normal Operation	909 days	04/24/07	04/19/11	7,026,975	-	-	-	-	-	1,384,000	1,738,125	1,688,125	1,638,125	578,600	-	-	-	-
1.3.1.2.2	FP-2 Retention in TF Particles- Normal Operation	909 days	01/15/09	01/09/13	6,808,350	-	-	-	-	-	-	-	1,708,975	1,738,125	1,688,125	1,673,125	-	-	-
1.3.1.3	Accident Condition Testing of DF and TF Particles	621 days	07/13/10	11/28/12	1,391,250	-	-	-	-	-	-	-	-	538,500	557,125	295,625	-	-	-
1.3.1.3.1	Accident Condition Testing of DF Particles	170 days	07/13/10	03/08/11	695,625	-	-	-	-	-	-	-	-	288,500	334,250	72,875	-	-	-
1.3.1.3.2	Accident Condition Testing of TF Particles	170 days	04/04/12	11/28/12	695,625	-	-	-	-	-	-	-	-	-	72,875	622,750	-	-	-
1.3.2	Retention in Fuel Compacts	1349 days	11/08/07	01/09/13	15,026,575	-	-	-	-	-	464,450	1,825,575	3,326,250	3,405,400	3,636,150	2,368,750	-	-	-
1.3.2.1	Retention in DF and TF Compacts	1349 days	11/08/07	01/09/13	13,635,325	-	-	-	-	-	464,450	1,825,575	3,326,250	3,405,400	2,940,525	1,673,125	-	-	-
1.3.2.1.1	FP-3 Retention in DF Compacts- Normal Operation	909 days	11/08/07	11/01/11	6,926,975	-	-	-	-	-	464,450	1,825,575	1,717,275	1,667,275	1,252,400	-	-	-	-
1.3.2.1.2	FP-4 Retention in TF Compacts- Normal Operation	909 days	01/14/09	01/09/13	6,708,350	-	-	-	-	-	-	-	1,608,975	1,738,125	1,688,125	1,673,125	-	-	-
1.3.2.2	Accident Condition Testing of DF and TF Compacts	481 days	01/26/11	11/28/12	1,391,250	-	-	-	-	-	-	-	-	-	695,625	695,625	-	-	-
1.3.2.2.1	Accident Condition Testing of DF Compacts	170 days	01/26/11	09/20/11	695,625	-	-	-	-	-	-	-	-	-	695,625	-	-	-	-
1.3.2.2.2	Accident Condition Testing of TF Compacts	170 days	04/05/12	11/28/12	695,625	-	-	-	-	-	-	-	-	-	-	695,625	-	-	-
1.4	Spent Fuel Disposal Technology	1691 days	12/30/05	06/25/12	6,807,380	-	-	-	-	1,469,200	969,200	969,200	969,200	969,200	969,200	492,180	-	-	-
1.4.1	Mechanical Integrity	1691 days	12/30/05	06/25/12	986,500	-	-	-	-	151,000	151,000	151,000	151,000	151,000	151,000	80,500	-	-	-
1.4.2	Long-Term Performance of Fuel Components	1691 days	12/30/05	06/25/12	4,476,800	-	-	-	-	1,027,200	627,200	627,200	627,200	627,200	627,200	313,600	-	-	-
1.4.2.1	Fuel Component Performance in Air	1691 days	01/02/06	06/25/12	2,238,400	-	-	-	-	513,600	313,600	313,600	313,600	313,600	313,600	156,800	-	-	-
1.4.2.2	Fuel Component Performance in Groundwater	1691 days	12/30/05	06/25/12	2,238,400	-	-	-	-	513,600	313,600	313,600	313,600	313,600	313,600	156,800	-	-	-
1.4.3	Radionuclide Transport	1691 days	12/30/05	06/25/12	1,344,080	-	-	-	-	291,000	191,000	191,000	191,000	191,000	191,000	98,080	-	-	-
1.4.3.1	Radionuclide Inventories in Spent Fuel	1691 days	12/30/05	06/25/12	642,040	-	-	-	-	140,500	90,500	90,500	90,500	90,500	90,500	49,040	-	-	-
1.4.3.2	Radionuclide Leaching Rates from Fuel Components	1691 days	12/30/05	06/25/12	702,040	-	-	-	-	150,500	100,500	100,500	100,500	100,500	100,500	49,040	-	-	-