PC-000510

Revision 0

Screening Tests for Selection of VHTR Advanced Fuel

Issued by General Atomics

for the Department of Energy

Contract No. DE-AC03-01SF22343

GENERAL ATOMICS

GENERAL ATOMICS

GA 2175 (1199E)

PROJECT CONTROL ISSUE SUMMARY

	DOC. CODE	PROJECT	DOCUMENT NO.	REV.
	TPN	30103	PC-000510	0

TITLE: Screening Tests for the Selection of VHTR Advanced Fuel

CM APPROVAL/ DATE	REV.	PREPARED BY	APPROVALS RESOURCE/ SUPPORT	APPROVALS PROJECT	REVISION DESCRIPTION/ W.O. NO.
Blayne Monen cn=Blayne Monen, o=Released, ou=Configuration Management, c=US 2003.12.10 12:30:36 -08'00' Signature Valid	0	D. L. Hanson D L Hanson	D. W. McEachern	A. S. Shenoy	Initial Issue W.O. 30103 300 11000

* SEE LIST OF EFFECTIVE PAGES

☐ GA PROPRIETARY INFORMATION

THIS DOCUMENT IS THE PROPERTY OF GENERAL ATOMICS. ANY TRANSMITTAL OF THIS DOCUMENT OUTSIDE GA WILL BE IN CONFIDENCE. EXCEPT WITH THE WRITTEN CONSENT OF GA, (1) THIS DOCUMENT MAY NOT BE COPIED IN WHOLE OR IN PART AND WILL BE RETURNED UPON REQUEST OR WHEN NO LONGER NEEDED BY RECIPIENT AND (2) INFORMATION CONTAINED HEREIN MAY NOT BE COMMUNICATED TO OTHERS AND MAY BE USED BY RECIPIENT ONLY FOR THE PURPOSE FOR WHICH IT WAS TRANSMITTED.

☒ NO GA PROPRIETARY INFORMATION

AAA 107 (08/2001)

LIST OF EFFECTIVE PAGES

Page Number	**Number of Pages**	**Revision**
i through ix	9	0
1-1 through 1-4	4	0
2-1 through 2-10	10	0
3-1 through 3-20	20	0
4-1 through 4-6	6	0
5-1 through 5-25	25	0
6-1 through 6-10	10	0
7-1 through 7-1	1	0
8-1 through 8-10	10	0
9-1 through 9-3	3	0
A-i through A-3	4	0
B-i through B-3	4	0
TOTAL NO. OF PAGES:	106	

Table of Contents

List of Tables

List of Figures

Acronyms and Abbreviations

Acronym	Definition
A&AE	activities and associated equipment
AF Plan	[Development Plan for Advanced High Temperature Coated-Particle Fuels]
AGR	Advanced Gas Reactor
ANL-W	Argonne National Laboratory – West
ATR	Advanced Test Reactor (INEEL)
BWXT	BWX Technologies [formerly B&W]
CCCTF	Core Conduction Cooldown Test Facility
COT	core outlet temperature
DB-MHR	Deep-Burn Modular Helium Reactor
DDN	Design Data Need
DTF	“designed-to-fail”
DOE	[United States] Department of Energy
EFPD	effective full-power days
EJ	engineering judgment
FIMA	fissions per initial metal atom (a measure of burnup in fuel)
FM	fission [product] metals
GA	General Atomics
GT-MHR	Gas-Turbine Modular Helium Reactor [prismatic core]
HEU	high-enriched uranium (usually ~93% U-235)
HFIR	High Flux Isotope Reactor [ORNL]
HHT	Hochtemperaturreaktor mit Helium Turbine [German direct-cycle HTR]
HTGR	High Temperature Gas-Cooled Reactor [generic term for reactor type]
HTR	High Temperature Reactor [pebble-bed core]
INEEL	Idaho National Engineering and Environmental Laboratory
IMGA	Irradiated Microsphere Gamma Analysis
IPyC	inner pyrolytic carbon coating in a TRISO particle
LEU	low-enriched uranium (<19.9% U-235)
LF	laser failed
LWR	light-water reactor

Acronym	**Definition**
MB	missing buffer
MHR	Modular Helium Reactor
NGNP	Next Generation Nuclear Plant
NNSA	National Nuclear Security Administration [independent agency within DOE]
NPR	New Production Reactor, an application for producing tritium
NTD	Near-Term Deployment
OPyC	outer pyrolytic carbon coating in a TRISO particle
ORNL	Oak Ridge National Laboratory
PBMR	Pebble Bed Modular Reactor
PGA	particle gas analyzer
PIE	post-irradiation examination
PIH	post-irradiation heating
P-PyC	"protective" pyrocarbon coating in a TRISO-P particle
QC	quality control
SEM	scanning electron microscope
R/B	release rate-to-birth rate ratio [fission gas release measurement]
RF	Russian Federation
RIAR	Research Institute of Atomic Reactors [Dimitrovgrad, RF]
RN	Radionuclide
SiC	silicon carbide coating in a TRISO particle
TBD	to be determined
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)
UCO	a mixture of UO_2 and UC_2
UO_2^*-B	UO_2-ZrC buffer
UO_2^*-C	UO_2-ZrC overcoat on kernel
VLPC	vented low pressure containment
WBS	work breakdown structure
WDX	wavelength dispersive X-ray [spectroscopy]
[-]	[provisional value]; subject to revision as the design evolves.

1. Executive Summary

This research and development plan describes a series of screening tests to provide the technical basis for selecting and qualifying an advanced coated-particle fuel for the VHTR. This screening plan is a subset of, and a precursor of, the umbrella Development Plan for Advanced High Temperature Coated-Particle Fuels which is proposed to satisfy the Design Data Needs for the VHTR in three related areas: (1) fuel process development, (2) fuel materials performance, and (3) fission product transport.

The possibilities for research and development into advanced coated-particle fuels are extensive; however, like all nuclear fuel R&D, the work is expensive and time consuming. Given these circumstances, the approach taken was to emphasize two advanced particle designs for which performance data have been published, suggesting that they may offer superior high temperature performance compared to conventional TRISO-coated fuel particles. The primary goal was to select and qualify an advanced particle design on a schedule consistent with the deployment schedule for a VHTR Demonstration Module (now referred as the Next Generation Nuclear Plant) which is projected to begin operation in early FY2016 at the INEEL.

In order to obtain early reviewer feedback on the direction and emphasis reflected in this screening plan, it was issued prior to the umbrella development plan of which it is an integral part. Nevertheless, these screening tests are intimately linked to the other elements of the umbrella plan (e.g., production of the test fuel is a fuel process development subtask); consequently, all elements of the umbrella program are shown on the schedules and included in the cost estimates contained herein. However, the descriptions of the planned fuel process development and the ex-core fission product transport tasks are presented in the umbrella plan and are not included here. In addition, the umbrella plan includes a technology status section which is also not included here.

The workscope in this screening plan includes: (1) capsule irradiation tests, (2) post-irradiation examinations, and (3) post-irradiation heating (accident simulation) tests which are proposed to identify, develop and qualify advanced coated-particle fuels capable of meeting anticipated VHTR fuel performance requirements.

It is assumed this advanced fuel program is an incremental program with the DOE-NE sponsored, Advanced Gas Reactor (AGR) fuel development program providing the base technology. A comprehensive list of fuel/fission product DDNs for the VHTR was first developed, and then the subset of these DDNs which would be addressed by this advanced fuel program was identified. To facilitate the earliest possible introduction of an advanced fuel, the strategy adopted is to place initial emphasis on UO_2^* (a conventional UO_2 kernel with a thin ZrC overcoat) and on “TRIZO”-coated (ZrC replacing SiC) UCO kernels; early screening tests will determine their adequacy for VHTR applications. Development of more “exotic” particle designs would follow as necessary.

Nine irradiation tests, using the multi-cell capsule being designed by the AGR program, and 35 post-irradiation heating tests were defined to satisfy these VHTR DDNs. The first two irradiation capsules and the first series of post-irradiation heating tests would be screening tests of TRISO-coated UO_2^* and of TRIZO-coated UCO. On the basis of these test results, one of these particles would be chosen as the reference advanced fuel particle, and subsequent tests would focus on qualifying this reference particle[1] and on validating the associated design methods for predicting its performance during normal plant operation and postulated accidents.

The summary schedule for the planned program is shown in Table 1-1. It is consistent with the overall goal of having a qualified advanced particle available at the time of the projected startup of a Demonstration VHTR Module in early FY2016. However, it is assumed at this writing that at least the first core for the Demonstration Module will use conventional TRISO-coated fuel. In other words, it is assumed that the AGR fuel program will demonstrate that conventional TRISO-coated UCO particles are adequate to meet VHTR performance requirements for operation at least with an 850 ^{o}C core outlet temperature (and, perhaps, to 1000 ^{o}C with core design changes). The durations of key tasks (e.g., capsule irradiation, post-irradiation examination, post-irradiation heating, etc.) were chosen to be consistent with the detailed estimates developed on the AGR program. The planned program continues into FY2016 to complete post-irradiation work on a planned screening capsule with more exotic coatings.

As summarized in Table 1-2, the total cost of the planned program is about $77 million. As with the task durations, the unit costs for key tasks (e.g., capsule irradiation, etc.) were chosen to be consistent with the detailed cost estimates developed on the AGR program. The cost estimates beyond FY2007 are highly speculative for the following reasons. With the current schedules, a number of key events are scheduled for completion by the end of FY2007. First, the preliminary design phase for the Demonstration Module will have been completed; consequently, the fuel performance requirements and service conditions will be much better established than at this writing. Secondly, the irradiation of the AGR-1 capsule with TRISO-coated UCO fuel will have been completed, giving a better indication of the performance potential of that fuel. Finally, the first two screening capsules planned under this program – VHTR-1 with TRISO-coated UO_2^* and VHTR–2 with TRIZO-coated UCO - will also have completed irradiation. At this point, it is anticipated that both the AGR fuel plan and this plan would be revisited and extensively revised (or, perhaps, even merged).

This 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 AGR fuel development program has been planned to develop and qualify LEU coated-particle fuel for use in future commercial HTGR designs, including the PBMR and GT-MHR. Secondly, 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. Coated-particle 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).

[1] In this Plan “reference” particle should be interpreted as shorthand for “reference advanced” fuel particle.

Table 1-1. Summary Schedule for Advanced Fuel Development

WBS	Task Name	Total Cost
1	**Fuel Design**	**$2,845,191**
1.1	**Design Data Needs**	**$224,231**
1.2	**Fuel Development Plan**	**$224,231**
1.3	**Fuel Specifications**	**$224,231**
1.4	**Model Development**	**$298,975**
1.5	**Design Methods Validation**	**$1,873,522**
2	**Fuel Development**	**$74,619,172**
2.1	**Fuel Process Development**	**$12,200,071**
2.2	**Fuel Materials Development**	**$47,425,524**
2.2.1	**Out-of-Pile Characterization**	**$299,961**
2.2.2	**Irradiation Testing**	**$14,710,714**
2.2.2.1	**Screening Tests**	**$6,817,370**
2.2.2.2	**Qualification Tests**	**$5,917,981**
2.2.2.3	**Validation Tests**	**$1,975,363**
2.2.3	**Postirradiation Examination**	**$8,199,432**
2.2.4	**Accident Simulation Tests**	**$24,215,418**
2.3	**Radionuclide Transport**	**$14,993,578**
2.3.1	**Transport in Reactor Core**	**$12,444,077**
2.3.1.1	**Normal Operation**	**$8,501,220**
2.3.1.2	**Accident Conditions**	**$3,942,857**
2.3.2	**Transport in Primary Circuit**	**$2,549,501**
2.3.2.1	**Normal Operation**	**$1,949,909**
2.3.2.2	**Accident Conditions**	**$599,593**

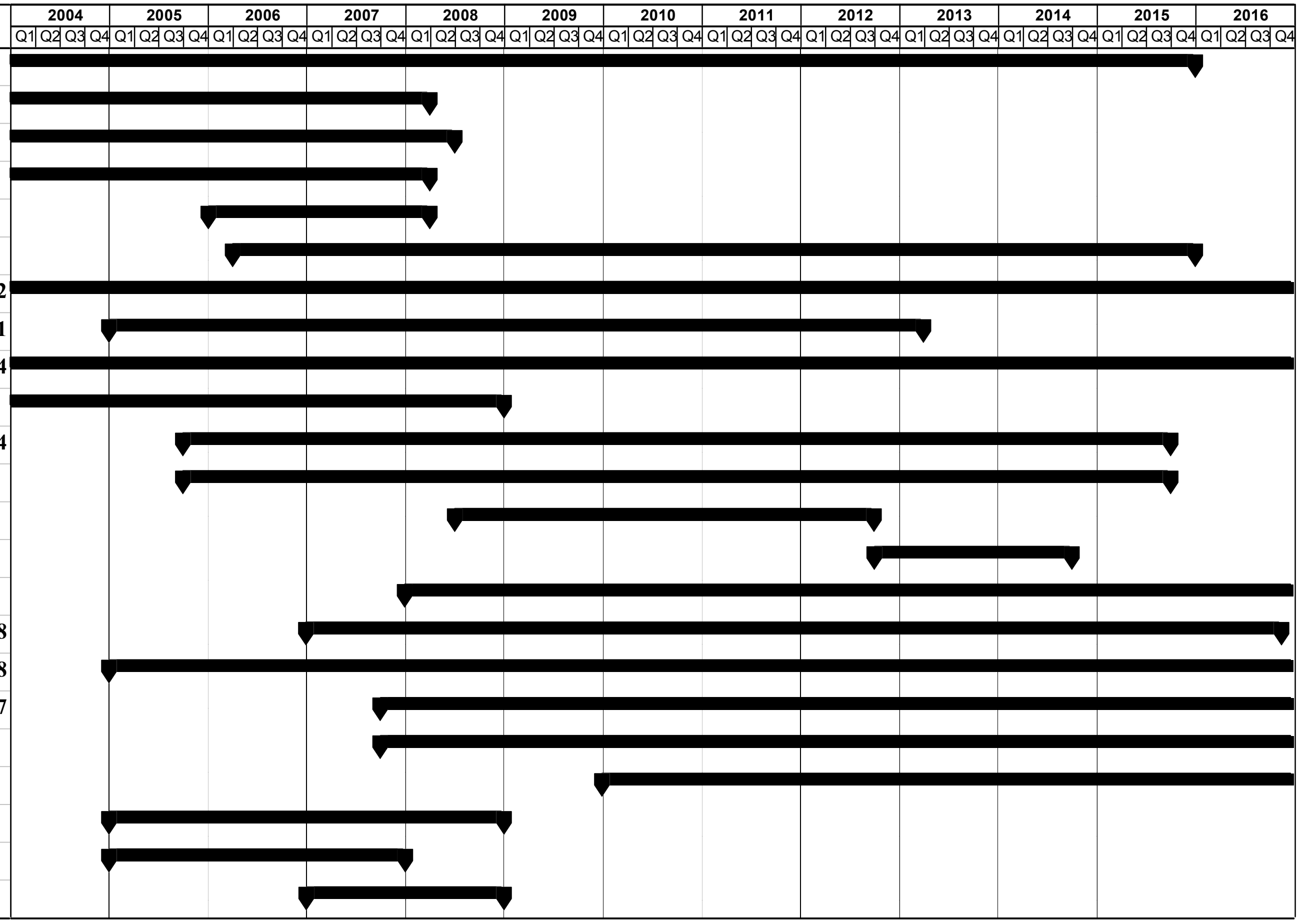


Table 1-2. Summary Cost Estimate for Advanced Fuel Development

Task	Annual Cost ($1000)													
	FY04	**FY05**	**FY06**	**FY07**	**FY08**	**FY09**	**FY10**	**FY11**	**FY12**	**FY13**	**FY14**	**FY15**	**FY16**	**Total**
Fuel Design														
Design Data Needs	75		75		75									225
Fuel Development Plan	75		75		75									225
Fuel Specifications	75		75		75									225
Model Development			148		150									298
Design Methods Validation			75							600	600	600		1875
Fuel Development														
Fuel Process Development		3557	2700	1499	1371	750	759	600	618	350				12204
Fuel Materials Development	60	60	60	60	60									300
Irradiation Testing														
Screening Tests		573	2272	1700						286	1136	850		6817
Qualification Tests					494	986	989	1972	1475					5916
Validation Tests									249	986	740			1975
Post-irradiation Examination					2326		293	870		2326		1220	1163	8198
Accident Simulation Tests				4000	2158	1281	620	2461	4154	2461	1854	3067	2158	24214
Radionuclide Transport														
Transport in Reactor Core														
Normal Operation				320	1387	1131	1943		249	986	740	1520	225	8501
Accident Conditions							2452					1392	99	3943
Transport in Primary Circuit														
Normal Operation		450	749	749										1948
Accident Conditions				300	300									600
Total	285	4640	6229	8628	8471	4148	7056	5903	6745	7995	5070	8649	3645	77464

2. Introduction

2.1 Purpose and Scope

This document contains the workscope, schedule and cost for a series of screening tests to provide the technical basis for selecting and qualifying advanced coated-particle fuels capable of meeting the anticipated Very High Temperature Reactor (VHTR) fuel performance requirements.

This screening plan is a subset of, and a precursor of, the umbrella Development Plan for Advanced High Temperature Coated-Particle Fuels (AF Plan 2003) which is proposed to satisfy the Design Data Needs (DDNs) for the VHTR in three related areas: (1) fuel process development, (2) fuel materials performance, and (3) fission product transport.

The possibilities for research and development into advanced coated-particle fuels are extensive; however, like all nuclear fuel R&D, the work is expensive and time consuming. Given these circumstances, the approach taken was to emphasize two advanced particle designs which may offer superior high temperature performance compared to conventional TRISO-coated fuel particles. The primary goal was to select and qualify an advanced particle design on a schedule consistent with the deployment schedule for the VHTR Demonstration Module which is projected to begin operation in early FY2016 at the Idaho National Engineering and Environmental Laboratory (INEEL).

In order to obtain early reviewer feedback on the direction and emphasis reflected in this screening plan, it was issued prior to the umbrella development plan of which it is an integral part. Nevertheless, these screening tests are intimately linked to the other elements of the umbrella plan (e.g., production of the test fuel is a fuel process development subtask); consequently, all elements of the umbrella program are shown on the schedules and included in the cost estimates included in this plan. However, the detailed descriptions of the planned fuel process development and the ex-core fission product transport tasks are presented in the umbrella plan and are not reproduced included here. In addition, the umbrella plan includes a technology status section which is also not included here.

2.2 Programmatic Overview

The programmatic context in which this screening plan was prepared is described in this subsection. (The narrative assumes that the reader has some knowledge of coated-particle fuels and their development history or is willing to consult the references.)

Advanced gas reactor (AGR) designs based upon High Temperature Gas-Cooled Reactor (HTGR) technology are capable of contributing to the resolution of key national and international issues. Among the Generation IV (Gen-IV) concepts, the VHTR is the nearest-term system capable of producing nuclear hydrogen and/or high-efficiency electricity (estimated to be deployable by 2020). Moreover, two gas-cooled reactors were identified by the complementary Near-Term Deployment (NTD) program as possibly being deployable within the next 10 years: the prismatic-core Gas Turbine-Modular Helium Reactor (GT-MHR) and the Pebble Bed Modular Reactor (PBMR). The GT-MHR is already being developed under a joint USDOE/MINATOM program for the purpose of destroying surplus Russian Federation (RF) weapons plutonium. Finally, the GT-MHR with a modified core design is also being evaluated as efficient burner of transuranic (TRU) materials. The primary benefit of the so-called Deep-Burn MHR (DB-MHR) would be to significantly reduce the long-term storage requirements for high-level waste generated by the currently operating nuclear reactors around the world.

The Generation IV project identified reactor system concepts for producing electricity, which excelled at meeting Generation IV goals related to safety, sustainability, proliferation resistance and physical security, and economics. One of these reactor system concepts, the VHTR is also uniquely suited for producing hydrogen without the consumption of fossil fuels or the emission of greenhouse gases. As a result DOE has selected this system for the Next Generation Nuclear Power (NGNP) Project,[2] a project to demonstrate emissions-free nuclear-assisted electricity and hydrogen production by 2015 (e.g., MacDonald 2003). A candidate NGNP design for producing nuclear hydrogen by either thermochemical water splitting or high-temperature electrolysis as well as electricity with a direct-cycle gas turbine is shown in Fig. 2-1 (Southworth 2003). A possible deployment schedule, assuming program initiation in early FY2004 is shown in Fig. 2-2 (Southworth 2003).

A hallmark philosophy of all modern HTGRs is to design the plant such that the radionuclides would be essentially retained in the core during normal operation and postulated accidents. The key to achieving this safety goal is the reliance on ceramic-coated fuel particles for primary fission product containment at their source, along with passive cooling to assure that the integrity of the coated particles is maintained even if the normal cooling systems were permanently disrupted. Consequently, these designs mandate the development and qualification of coated-particle fuels that meet stringent requirements for as-manufactured quality and in-service coating integrity even for beyond design-basis accidents

The Germans successfully produced and demonstrated high performance TRISO-coated fuel for their pebble-bed High Temperature Reactor (HTR) designs in the 1980s. No US-manufactured coated particle has exhibited equivalent performance to date. More generically, the service conditions proposed for the advanced applications introduced above are more demanding than those associated with the German steam-cycle HTR designs of the 1980s (e.g., 10% FIMA burnup, 700 ^{o}C core outlet temperature, etc.). In particular, the VHTR preconceptual designs are characterized by significantly higher burnups (>20% FIMA) and much higher core outlet temperatures (850 – 1000 ^{o}C). The plutonium-burning GT-MHR and the TRU-burning DB-MHR are characterized by much higher burnups (>70% FIMA) and significantly higher core temperatures (~850 ^{o}C). Consequently, fuel development and qualification were identified as essential early technology development needs to assure concept viability for each of the aforementioned advanced designs; as a result, a series of fuel development plans have been or are being prepared at this writing as discussed below.

The Technical Program Plan for the Advanced Gas Reactor Fuel Development and Qualification Program (AGR Plan 2003) has the overall goals of (1) providing a qualified fuel to support the design and licensing of the Gen-IV VHTR, and (2) supporting near-term deployment of an AGR for commercial energy production. The AGR fuel program will focus on developing and

[2] The terminology "Freedom Power Project" was used earlier (e.g., Magwood 2003); the more generic term "VHTR Demonstration Module" and "NGNP" are used interchangeably in this Plan.

qualifying TRISO-coated (SiC) fuel with a low-enriched (LEU) UCO kernel in support of both the VHTR and NTD programs. This particle could be used in either a prismatic or a pebble-bed core; however, as presently conceived, the AGR program will utilize cylindrical fuel compacts characteristic of prismatic cores. Complementary development of the reference German fuel – LEU TRISO UO_2 in fuel spheres – is on-going in South Africa, China, and Europe.

The Russian Fuel Development Plan for the International GT-MHR (RF Plan 2002) has the goal of developing and qualifying high-burnup, TRISO-coated $PuO_{1.68}$ fuel which uses weapons-grade Pu as the feedstock (McEachern/Makarov 2001). The reactor design for burning this surplus RF weapons Pu is a 600 MW(t) direct-cycle GT-MHR with a core outlet temperature of 850 ^oC (OKBM 1997). The emphasis is on achieving maximum Pu-239 destruction in a single pass. Based upon previous irradiation tests with TRISO-coated PuO_x particles (e.g., Miller 1985), it should be possible to meet the fuel performance requirements with a conventional TRISO (SiC) coating system. The RF fuel program will include as a backup a particle design that uses a PuO_x kernel diluted with Zr or C to lower the effective burnup. The RF Plan is in draft form at this writing. It is anticipated that the final program will be similar to Fuel Development Plan for the Plutonium Consumption-Modular Helium Reactor (Turner 1994) which had a similar mission (but with surplus US weapons Pu), plant design, and fuel design.

The Deep-Burn Modular Helium Reactor Fuel Development Plan (DB-MHR Plan 2002) has the goal of developing and qualifying a fuel system for a thermal transmutation burner (Venneri 2001). 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 TRISO-coated, 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 DF are combined and made into a TRISO-coated Transmutation Fuel (TF). The reactor design for burning this is a 600 MW(t) direct-cycle GT-MHR with a modified fuel element and a core outlet temperature of 850 ^oC. The emphasis is on achieving high-burnup in a single pass. Based upon successful past irradiations of high-burnup TRISO-coated fuel particles, the Plan assumes that a conventional TRISO-coating (SiC) system will meet fuel requirements; however, the Plan does include provisions for switching to a TRIZO-coating (ZrC) system should early screening tests demonstrate that conventional TRISO coatings are inadequate for DF and/or TF particles

The above fuel development plans emphasize coated-particle designs with fuel kernels custom tailored for the specific application but with conventional TRISO (SiC) coating systems. The extensive international experience with a large variety of TRISO-coated fuel particles (e.g., IAEA 1997) strongly indicates that SiC-based coating systems should prove adequate for a broad range of AGR applications with core outlet temperatures of at least 850 ^oC and, perhaps, up to 1000 ^oC (with certain core design changes to limit fuel temperatures). However, as core outlet temperatures are increased to 1000 ^oC and higher, the ultimate performance limits of SiC-based, conventional TRISO coatings will be reached at some point. In recognition of this eventuality, the Advanced Fuel Cycle Initiative (AFCI) has sponsored the preparation of a Development Plan for Advanced High Temperature Coated-Particle Fuels (AF Plan 2003) which has the overall goal of identifying, screening, selecting, and qualifying advanced coated-particle designs with significantly higher temperature capabilities than conventional TRISO particles.

A number of candidate, advanced coated-particle designs have been explored which appear to promise superior high temperature performance compared to conventional TRISO particles. Typically, these advanced particle designs have been fabricated in small quantities in laboratory-scale equipment and subjected to varying degrees of exploratory testing, including out-of-pile tests, irradiation tests, post-irradiation examination (PIE), and post-irradiation heating (PIH) tests. As summarized in the AF Plan, two promising advanced particle designs appear to be more mature than the others (at least based upon information published in the open literature): (1) TRISO-coated UO_2^* (conventional UO_2 kernel with a thin ZrC overcoat) and of TRIZO-coated (ZrC replacing SiC) UCO; the available data on UO_2^* and on ZrC coatings have been reviewed previously (e.g., in Section 7 of IAEA 1997). Consequently, the strategy adopted in the Advanced Fuel Plan is to place initial emphasis on UO_2^* and ZrC development; early screening tests will determine their adequacy for VHTR applications. Development of more "exotic" particle designs would follow as necessary (AF Plan 2003).

As stated above, this screening plan is a subset of the Advanced Fuel Plan. This plan focuses on the irradiation testing, post-irradiation examination, and post-irradiation heating tests necessary to select and qualify an advanced fuel which capable of meeting the anticipated VHTR fuel performance requirements. As a result, the emphasis is on those DDNs related to fuel materials development and to fission product release from the fuel particles. The DDNs related fuel process development and the remaining, ex-core fission product transport DDNs are addressed in the umbrella AF Plan.

Coated-particle fuel development is expensive and time consuming; consequently, it is impractical to systematically investigate all promising advanced designs. Thus, a considerable degree of engineering judgment had to be exercised in developing the test matrices presented herein. In the present circumstance, that judgment is strongly tempered by past fuel development experience which indicates that is unwise to make multiple simultaneous changes in the particle design. Experience also indicates that is essential to get early irradiation and post-irradiation heating data before the effects of particle design changes can be reliably determined.

A relevant example of past experience is the TRISO-P particle (Leikind 1993) which was adopted as the reference particle for gas-cooled New Production Reactor. The TRISO-P design featured both a significantly thicker and denser inner pyrocarbon (IPyC) layer and an added porous "protective" (P-PyC) outer layer. Both design changes were made to solve perceived problems compact fabrication. The IPyC layer was thickened to improve the quality of SiC coating by reducing the potential for producing defects during deposition of the SiC coating. The outer P-PyC layer was added to reduce the potential for introducing SiC defects from particle-to-particle contact during compacting. The design changes resolved these process issues, and the as-manufactured quality of the fuel compacts was dramatically improved. However, under irradiation the thicker (and more anisotropic) IPyC developed radial cracks which served as stress risers in the SiC layer, and the porous P-PyC layers shrank excessively and developed cracks that propagated into the OPyC layer, causing a high fraction of the OPyC layers to fail. The combined result of these design "improvements" was an order-of-magnitude increase in the in-service failure rates compared to that of conventional US-made TRISO particles even though the as-manufactured quality had been much improved.

Given this experience and perspective, it should not be surprising that the two leading advanced fuel designs represent incremental changes in the conventional FRG and US particle designs, respectively. The UO_2^* particle, of which there are two variants, is essentially a modification of the standard FRG TRISO-coated UO_2 particle. The only design change is the addition of ZrC to the particle: either as a thin ZrC coating applied over a thin PyC seal coat on the UO_2 kernel (referred to as UO_2^*-C herein) or codeposited with the porous PyC buffer layer (referred to as UO_2^*-B herein). As discussed in the "Status" section of the Advanced Fuel Plan, UO_2^* particles, especially the UO_2^*-C variant, appear to perform far better than conventional TRISO-coated particles (e.g., Ag-110m is completely retained at 1500 °C for 10,000 hours). The TRIZO particle is the standard LEU UCO particle with the SiC coating replaced by a ZrC coating. Again as discussed in the "Status" section of the Advanced Fuel Plan, ZrC coatings are more thermally stable than SiC and are not degraded by palladium attack at high temperatures (>~1400 °C).

Moreover, it also should not be surprising that this screening plan emphasizes obtaining early irradiation and post-irradiation heating data to determine the performance limits of these advanced designs as soon as practical. To that end, the plan accepts the risk of performing the initial screening tests with particles that have been fabricated using published process conditions and laboratory-scale equipment and of delaying significant process optimization studies until a reference particle has been selected.

2.3 Key Assumptions/Development Strategy

The VHTR and AFCI programs are both at an early stage of definition; hence, there are many significant technical and programmatic uncertainties at this writing. This circumstance mandated that a number of key assumptions be made and a development strategy formulated before this plan could be drafted. As the program definitions mature and the attendant uncertainties are reduced, some of these assumptions may be invalidated, and the development strategy may have to be modified. Likewise, as early test data are obtained, further revisions may be appropriate. With these caveats, the basis for the AF Plan is summarized below:

- ♦ As a point of departure, the fuel performance requirements for the VHTR with a 1000 °C core outlet temperature will be assumed to be the same as those for the direct-cycle GT-MHR with an 850 °C core outlet temperature. This assumption may prove to be too ambitious; in particular, the allowable core metal release limits (Ag, Cs, etc.) may have to be increased even if the failure limits are maintained because of the higher fuel and graphite temperatures.
- ♦ Improvements will be required in the fuel particle design, in the fuel-element design, and in the core design to limit fuel temperatures during normal operation and core heatup accidents in order to meet VHTR fuel performance requirements[3] with a 1000 °C core outlet temperature.

[3] "Performance requirements" here refer to limits on in-service coating failure and fission product release from the core during normal operation and accidents.

- Fuel particle designs investigated in this program will be suitable for use in both prismatic and pebble-bed cores.
- The fuel cycle will be based upon 20% enriched LEU, will achieve high fissile material utilization, and will be closed by direct disposal of unprocessed spent fuel elements.
- This program is an incremental program; the DOE AGR program will provide the base technology including fission product transport in core graphite and transport ex-core.
- The requisite experimental facilities to fabricate and to test advanced fuels will be available on the required schedule.
- Relevant international data will be acquired, analyzed and used as applicable.
- All operational test facilities, including foreign test facilities, can be utilized.
- The choice of test facilities will be based upon: (1) its ability to meet test specifications and (2) total cost.
- Candidate fuel particle designs will be irradiated as loose particles and in cylindrical fuel compacts (i.e., no production or irradiation of pebbles is anticipated).

2.4 Program Coordination and Collaboration

The VHTR 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 DOE-NE sponsored, AGR fuel development program has been planned to develop and qualify LEU TRISO-coated fuel to be used in commercial PBMR and GT-MHR designs. Secondly, the joint DOE-NNSA/MINATOM International GT-MHR program (OKBM 1997) for the disposition of surplus Russian weapons plutonium is developing high-burnup, TRISO-coated Pu fuel. Fuel development activities sponsored by the Europe Union, China, and Japan should also produce directly relevant data (e.g., the latter's planned development of advanced ZrC coatings).

2.5 Plan Organization and Content

This screening plan is organized into 10 sections. Section 1 provides a summary of the most important features of the plan, including cost and schedule information.

Section 2 provides a programmatic context and briefly describes the VHTR concept and the plan to develop high temperature, coated-particle fuel for it.

Section 3 presents the fuel performance requirements in terms of as-manufactured quality and performance of the coatings under irradiation and accident conditions. These requirements are provisional because the design of the VHTR is at an early stage. However, the requirements are presented in this preliminary form to guide the initial work on the fuel development. Section 3 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 4 describes the Design Data Needs to: (1) fabricate the coated particle fuel, (2) predict its performance in the reactor core, and (3) predict the radionuclide transport throughout the VHTR plant.

The heart of this screening plan is Section 5, which describes the development activities proposed to satisfy the DDNs presented in Section 4. The development scope is divided into Fuel Materials Development and Radionuclide Transport.

Section 6 describes the facilities to be used. Generally, this program will utilize existing US facilities. While some facilities will need to be modified, there is no requirement for major new facilities.

Section 7 states the Quality Assurance requirements that will be applied to the development activities.

Section 8 presents costs and schedule, organized by a simple Work Breakdown Structure (WBS).

Section 9 presents a list of the types of deliverables that will be prepared in each of the WBS elements.

2.6 References for Section 2

[AF Plan] "Development Plan for Advanced High Temperature Coated-Particle Fuels," GA Document PC-000XXX/0, General Atomics, December 2003 (In preparation).

[AGR Plan] "Technical Program Plan for the Advanced Gas Reactor Fuel Development and Qualification Program" ORNL/TM-2002/262, Oak Ridge National Laboratory, April 2003.

[DB-MHR Plan] "Deep-Burn Modular Helium Reactor Fuel Development Plan," GA-224-0-TRT-000167, General Atomics, September 2002.

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

Leikind, B., et al., "MHTGR TRISO-P Fuel Failure Evaluation Report," DOE-HTGR-90390, CEGA, October 1993.

MacDonald, P. E., et al., "Next Generation Nuclear Plant Project – A Demonstration Of Nuclear-Assisted Hydrogen And Electricity Production Using Advanced High-Temperature Gas-Cooled Reactor Technology: Preliminary Preconceptual Point Design Report," INEEL/EXT-03-00870, INEEL, July 2003.

McEachern, D., V. Makarov and I. Suslov, "The International GT-MHR Fuel Program," International HTR Fuel Seminar, Brussels, Belgium, February 2001.

Miller, C. M., and W. J. Scheffel, "Post-irradiation Examination and Evaluation of Peach Bottom FTE-13," General Atomics Report 906939, November 1985.

OKBM, "Conceptual Design Report," Int. GT-MHR 10025, Rev. 1, November, 1997.

[RF Plan 2002] "Fuel Development Plan," Product No. 03.0102, Russian Research Center-Kurchatov Institute, July 2002 (Draft).

Southworth, F. H., "Very High Temperature Gas Cooled Reactor Demonstrator…," Presentation at Gen-IV Steering Committee Meeting, INEEL, June 17, 2003

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

Venneri, F., et al., "Deep Burn Transmutation," General Atomics Report No. FDO-EDO-N-TRT-X-000132, December 2001.

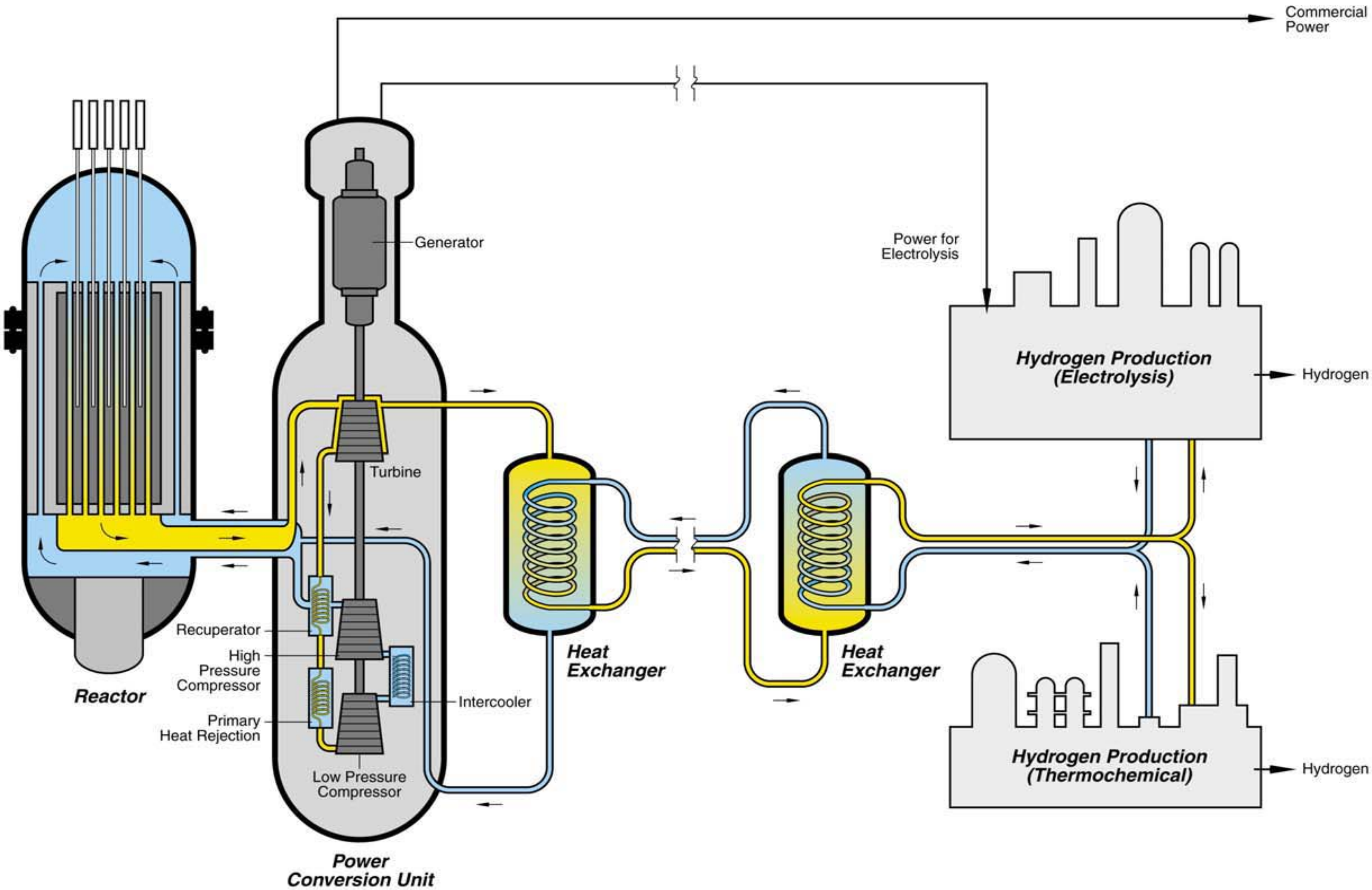


Figure 2-1. VHTR Demonstration Module

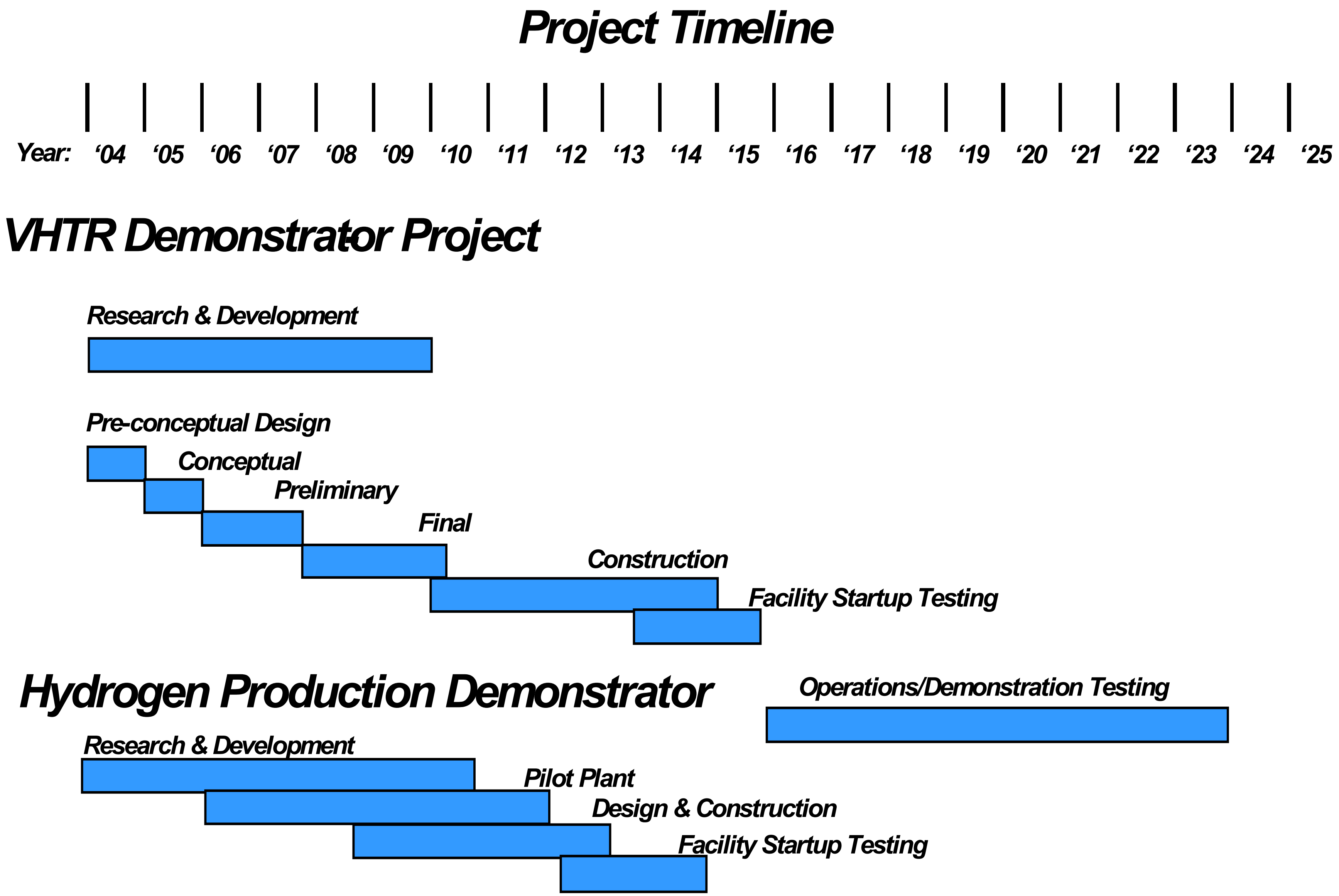


Figure 2-2. VHTR Demonstration Module Master Schedule

3. Provisional Fuel Requirements

3.1 VHTR Fuel Description

The VHTR is a graphite-moderated, helium-cooled reactor designed for high efficiency operation and for passive safety. The VHTR produces high temperature helium capable of driving a gas turbine that can turn an electrical generator and/or providing nuclear process heat for a broad spectrum of energy-intensive, high-temperature applications, including hydrogen production (e.g., MacDonald 2003a). Passive safety is possible because of the high heat capacity provided in the core by the graphite fuel elements, and the ability of the coated fuel particles and the ceramic core to maintain their integrity at high temperature.

The VHTR plant is still in the early definition phase; however, it is anticipated that the Reactor System for the VHTR will be similar to that for the direct-cycle GT-MHR with the likely exception that the core operating temperatures will be higher. With that anticipation, the GT-MHR reactor core and fuel design (Shenoy 1996) are briefly described below.

3.1.1 Physical Description

The standard direct-cycle GT-MHR plant is comprised of four 600 MW(t) modules which generate a total of 1148 MW(e). The module components are contained within three steel pressure vessels: reactor system vessel, power conversion system vessel, and cross vessel. All three vessels are sited underground in a concrete silo, which serves as an independent, vented low pressure containment (VLPC) structure.

The GT-MHR core, located inside the reactor vessel, is designed to produce 600 MW(t) at a power density of 6.6 W/cm^3. The active core consists of an assembly of fuel elements in the form of hexagonal graphite blocks containing nuclear fuel compacts and coolant channels. The fuel elements are stacked 10 high in the core to form columns that rest on graphite support structures. As shown in Fig. 3-1, the active core is composed of 102 fuel columns in an annular arrangement. The annular core configuration was adopted to achieve maximum power rating and still permit passive core heat removal while maintaining the peak fuel temperature below 1600 ^{o}C during the worst case accident condition of total loss of coolant and loss of flow, thereby assuring that fuel integrity is not impaired. Some key core attributes are summarized in Table 3-1. The GT-MHR fuel element and its components are shown in Fig. 3-2. The following subsections provide brief descriptions of the coated fuel particles, fuel compacts, and fuel-element graphite blocks.

3.1.1.1 Fuel Particles

The reference fuel for the GT-MHR consists of microspheres of uranium oxycarbide that are coated with multiple layers of pyrocarbon and silicon carbide. The GT-MHR core is designed to use a blend of two different particle types: a fissile particle that is enriched to 19.8% U-235 and fertile particle with natural uranium (0.7% U-235). The fissile/fertile loading ratio is varied with location in the core, in order to optimize reactivity control, minimize power peaking, and maximize fuel burnup. The buffer, inner pyrolytic carbon (IPyC), silicon carbide (SiC), and outer pyrolytic carbon (OPyC) layers are referred to collectively as a TRISO[4] coating. The coating system can be viewed as a miniature multi-shell pressure vessel that provides containment of radionuclides and gases. This coating system is also an excellent engineered barrier for long-term retention of radionuclides in a repository environment. The reference TRISO particle design parameters are given in Table 3-2 (advanced coated-particle designs are described in Section 3.3).

3.1.1.2 Fuel Compacts

Each fuel compact is a mixture of fissile and fertile particles bonded together with a carbonaceous matrix into a cylindrical-shaped compact with dimensions 12.45 mm (0.49 in.) in diameter and 49.3 mm (1.94 in.) in length. The compact matrix material will be based upon a thermosetting resin similar to that used in the fabrication of spherical fuel elements for pebble-bed reactors. The fuel compacts are stacked in the blind fuel holes of the graphite fuel element. Graphite plugs are cemented into the tops of the fuel holes to enclose the stacked compacts. Because of sorption mechanisms, the fuel compacts can provide an additional barrier to the release of metallic fission products. Compact design parameters are given in Table 3-3.

3.1.1.3 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 360 mm (14.172 in.) across flats and 793 mm (3.122 in.) 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 15.88 mm (0.625 in.) in diameter; fuel holes are 12.7 mm (0.5 in.) in diameter. Each block has approximately two fuel holes per coolant hole, located on a triangular pitch of 18.8 mm (0.740 in.) 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.

3.1.2 Fuel Cycle

It is anticipated that the fuel cycle adopted for the VHTR Demonstration Module will be similar or the same as the reference cycle for the electricity-producing GT-MHR (the duty cycles will be different assuming that the former will produce hydrogen as well as electricity). For the equilibrium GT-MHR fuel cycle, one-half of the core (510 fuel elements) is reloaded every 417 effective full-power days (EFPD), corresponding to an equilibrium residence time of 834 EFPD for each fuel element. Each reload segment contains 1746 kg of low-enriched uranium and 507 kg of natural uranium. With a capacity factor of 85%, the GT-MHR would discharge 510 fuel elements every 16 months, or an average of about 380 elements per calendar year. Over its 60-yr plant life, a single GT-MHR module would discharge a total of about 23,000 spent-fuel elements.

---

[4] TRISO is an acronym for TRI-material, ISOtropic, with the materials being low-density pyrolytic carbon (buffer), high density pyrolytic carbon (IPyC and OPyC), and SiC.

3.1.3 VHTR Service Conditions

Peak service conditions for VHTR fuel are assumed here that are consistent with previous core designs with outlet temperatures of 850 °C and higher. They are subject to revision when the conceptual and preliminary core designs are completed for a prismatic-core VHTR. These fuel service conditions are intended to enable the VHTR achieve its goals of nuclear hydrogen production and high-efficiency electricity generation. These assumed VHTR service conditions are compared with the conditions for the 850 °C GT-MHR in Table 3-4 (Sherman 1995).

The peak fuel temperature in the commercial GT-MHR with an 850 °C core outlet temperature is expected to be ~1250 °C for normal operation and <1600 °C for depressurized core heatup accidents. A design goal for the VHTR is optimize the core and plant design such that these peak temperature limits can also be met (or nearly so) with a 1000 °C core outlet temperature. Core design changes will permit increased core outlet temperatures without a proportionate increase in peak fuel temperatures during normal operation although some increases in the average fuel and graphite temperatures should be expected since the average temperatures largely track the coolant temperatures. Design changes to the reference 600 MW(t) GT-MHR core (Sherman 1995) have been identified which have significant potential for accommodating higher core outlet temperatures; they include fuel shuffling schemes, fixed column orifices, and fuel-element redesigns (e.g., MacDonald 2003b). For core heatup accidents, a 150 °C increase in core outlet temperature translates into about a 50 °C increase in peak fuel temperature (e.g., MacDonald 2003b).

Given the above, there is good reason to believe that design optimization and evolution will produce a core design that will permit the use of conventional TRISO-coated particles in a 600 MW(t) VHTR with a 1000 °C outlet temperature. Nevertheless, an advanced coated-particle fuel with higher temperature capabilities is highly desirable to facilitate higher core outlet temperatures and higher power levels which is the primary motivation for this plan.

These provisional service conditions are needed as an initial guide to the fuel development and the reactor core design. Bounding conditions are needed to perform fuel-particle design analyses, to prepare provisional fuel product specifications, and to plan the details of the fuel irradiation and testing programs. Core designers need this information to guide them in the trade studies required to optimize the core design. Since certain coating failure mechanisms depend on the exact history of time, temperature, burnup, and fast neutron fluence, it may be necessary to define more detailed limits for combinations of these core and fuel cycle parameters as part of the overall VHTR design and development effort.

3.2 VHTR Fuel Requirements

Like all Modular Helium Reactors, the radionuclide containment system for the VHTR will be comprised of multiple barriers to limit radionuclide release from the core to the environment to insignificant levels during normal operation and a 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 structural graphite, (4) the primary coolant pressure boundary; and (5) the Vented Low-Pressure Confinement building. As part of the design process, performance requirements must be derived for each of these release barriers.

Of these multiple release barriers, the particle coatings are the most important. Moreover, the in-reactor performance characteristics of the coated-particle fuel are strongly influenced by its as-manufactured attributes. Consequently, the fuel performance requirements and fuel quality requirements (allowable, as-manufactured, heavy-metal contamination and coating defects) must be systematically defined and controlled. Traditionally, the as-manufactured fuel attributes are controlled by a combination of fuel product- and fuel process specifications.

The logic for deriving these fuel quality specifications is illustrated in Fig. 3-3 (Hanson 2001). Top-level requirements for the VHTR will be defined by both the regulators and the user. Lower-level requirements will then be systematically derived using a top-down functional analysis methodology. With this approach, the radionuclide control requirements for each of the release barriers can be defined. For example, starting with the allowable doses at the site boundary, limits on Curie releases from the plant, from the VLPC, from the reactor vessel, and from the reactor core will be successively derived. Fuel failure criteria are in turn derived from the allowable core release limits. Finally, the required as-manufactured fuel attributes will be derived from the in-reactor fuel failure criteria providing a logical basis for the fuel quality specifications.

In-service fuel performance requirements and as-manufactured fuel quality requirements have not yet been defined for a generic VHTR or for the VHTR Demonstration Module. The fuel performance and quality requirements adopted for a given HTGR design along with the fuel service conditions will determine the amount of technology development that will be necessary to support the design and license the plant. Consequently, it is critically important that a comprehensive set of fuel requirements be derived for the VHTR early in the design process.

As a point of departure for preparing this screening plan, the fuel requirements for the VHTR with a 1000 ^{o}C core outlet temperature were assumed to be the same as those for the direct-cycle GT-MHR with an 850 ^{o}C core outlet temperature (Munoz 1994). This assumption may prove to be too ambitious. It is reasonable to expect that these as-manufactured fuel quality limits can be met since the Germans met or exceeded comparable limits in the late 1970s (e.g., Hanson 2001). However, the in-service fuel performance limits could prove problematic; in particular, the allowable core metal release limits (Ag, Cs, etc.) may have to be increased even if the failure limits are maintained because of the higher average core temperatures which will result in less retention by the fuel kernels of failed particles and by the fuel-element graphite.

The provisional VHTR fuel performance and quality requirements are summarized in Table 3-5, and the provisional metal release limits are shown in Table 3-6. For perspective, the allowable metal release limits for the US steam-cycle MHTGR plant and for the German direct-cycle HHT plant are also shown in the latter table (Hanson 1995). The limits on volatile metal release are particularly speculative at this writing (because they were developed for a direct-cycle GT-MHR rather than for a VHTR), and considerable plant design and fuel development will likely be required to optimize them.

3.3 VHTR Fuel Product Specifications

Ceramic-coated fuel particles used in HTGRs are designed to retain radionuclides are their source during normal operation and postulated accidents; the fuel performance and quality requirements anticipated for the VHTR were summarized in the previous subsection. The fuel requirements for the VHTR will be formalized and controlled by the fuel product and process specifications.

As indicated in Fig. 3-4 for a conventional TRISO particle, the coating layers have specialized purposes but, in composite, provide a high-integrity pressure vessel which is extremely retentive of fission products (e.g., Bullock, 1994). The purpose of the buffer layer is to provide a reservoir for fission gases released from the fuel kernel and to attenuate fission recoils. The main purposes of the IPyC are to provide a smooth regular substrate for the deposition of a high-integrity SiC coating and to prevent Cl_2 and HCl from permeating the fuel kernel during the SiC deposition process; hence, a major benefit of the IPyC coating is realized during fuel fabrication. The IPyC coating, which shrinks under irradiation, also produces a compressive stress in the dimensionally stable SiC (assuming that the former is mechanically attached to the latter). If the IPyC cracks radially or, perhaps, if it partially debonds from the SiC, it can cause high local stresses at the SiC inner surface and subsequent SiC failure as observed in the TRISO-P particles (Leikind 1993).

The most important coating in a TRISO particle is the SiC which provides most of the structural strength and dimensional stability and which serves as the primary barrier to the release of fission products, particularly the metallic fission products. Like the IPyC coating, the OPyC coating shrinks under irradiation, and both produce compressive stresses in the SiC which compensate for the tensile stress in the SiC induced by the internal gas pressure (in fact, TRISO particles are typically designed so that SiC layer never goes into tension). PyC coatings have also been shown to effectively retain fission gases in fuel particles with defective or failed SiC layers up to about 1800 ^{o}C.

Coated fuel particles must be designed and specified to maintain a high degree of coating integrity during normal operation and postulated accidents. After decades of international coated-particle fuel development, a number of potential failure mechanisms have been identified that can challenge coating integrity (e.g., IAEA-TECDOC-978 1997). Candidate fuel particle designs for the VHTR must be demonstrated by test to be sufficiently resistant to these failure mechanisms to meet the coating integrity requirements summarized in the previous subsection. Prior to actual operation of a first VHTR module, analytical design methods must be used to predict in-core fuel performance and to demonstrate compliance with fuel performance and fission product release criteria.

The current design methods used to predict fuel performance in prismatic-core HTGRs (e.g., IAEA-TECDOC-978 1997) consider eight potential failure mechanisms which are illustrated schematically in Fig. 3-5:

1. Coating damage during fuel manufacture, resulting in heavy metal contamination.

2. Pressure-induced failure in particles with defective or missing coating layers.

3. Pressure-induced failure in standard particles, i.e., particles without manufacturing defects.

4. Irradiation-induced failure of the OPyC coating.

5. Heavy-metal dispersion during SiC coating deposition and subsequent accelerated SiC corrosion during irradiation.

6. Failure of the SiC coating due to kernel migration in the presence of a thermal gradient.

7. Failure of the SiC coating caused by fission product/SiC interaction.

8. Failure of the SiC coating by thermal decomposition.

The conventional, TRISO-coated, fissile and fertile particle designs specified for the GT-MHR (Munoz 1994) and summarized in Table 3-2 should be capable of meeting anticipated VHTR fuel requirements at least with a core outlet temperature of 850 °C. However, as core outlet temperatures are increased to >~950 °C, the ultimate performance limits of SiC-based conventional TRISO coatings will be reached; hence, the rationale for this development plan. As already introduced, two promising advanced particle designs – UO_2^* and TRIZO - appear to be more mature than the others and, hence, will be investigated here.

Provisional product specifications for UO_2^* and TRIZO particles are given in Tables 3-7 and 3-8, respectively. These specifications are proposed as a starting point for use in core design, fuel particle design, and process development tasks. It is anticipated that these specifications will need to be revised and embellished a number of times, certainly after the screening and qualification test phases are completed.

3.4 References for Section 3

Bullock, R. E., and C. Young, "TRISO Fuel Particle Design Basis," GA No. DOE-GT-MHR-100225, September 1994.

Hanson, D. L., "Revised Core Release Criteria for Commercial GT-MHR," GA Internal Memo 818:DLH:143:94, August 1, 1994.

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

Homan, F., T. Lindemer, E. Long, T. Tiegs, and R. Beatty, "Stiochiometric Effects on Performance of High-Temperature Gas-Cooled Reactor Fuels from the U-C-O System," *Nucl. Tech*, **35**, pp. 428-441, 1977.

Leikind, B., et al., "MHTGR TRISO-P Fuel Failure Evaluation Report," DOE-HTGR-90390, CEGA, October 1993.

MacDonald, P. E., et al., "Next Generation Nuclear Plant Project – A Demonstration Of Nuclear-Assisted Hydrogen And Electricity Production Using Advanced High-Temperature Gas-Cooled Reactor Technology: Preliminary Preconceptual Point Design Report," INEEL/EXT-03-00870, INEEL, July 2003.

MacDonald, P. E., et al., "NGNP Preliminary Point Design – Results of the Initial Neutronics and Thermal-Hydraulic Assessments," INEEL/EXT-03-00870, Rev. 1, INEEL, September 2003

Miller, C. M, and W. J. Scheffel, "Post-irradiation Examination and Evaluation of Peach Bottom FTE-13," General Atomics Report 906939, November 1985.

Munoz, S. P., “Fuel Product Specification [for the GT-MHR],” DOE-GT-MHR-100209, Rev. 0, May 1994.

Shenoy, A., “Gas Turbine-Modular Helium Reactor (GT-MHR) Conceptual Design Description Report,” GA Report 910720, Rev. 1, General Atomics, July 1996.

Sherman, R., “3D Rodded Burnup Results for the GT-MHR, “ GA Document 910832, Rev. 0, General Atomics, July 6, 1995.

Table 3-1. GT-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 (EFPD)	425
Number of Fuel Columns	102
Number of Fuel Elements	1020
Reactor Arrangement	Annular - Three Rings of Fuel Columns
Power Density (kW/m^3)	6.6
Coolant	Helium
Coolant Pressure	1025 psi (7 MPa)
Average Outlet Gas Temperature (°C)	490
Average Outlet Gas Temperature (°C)	850

Table 3-2. TRISO Particle Design Parameters

Parameter	Fissile Particle	Fertile Particle
Composition	$UC_{0.5}O_{1.5}$	$UC_{0.5}O_{1.5}$
Uranium enrichment, %	19.8	0.7 (Natural Uranium)
Design burnup (% FIMA)	26	7
Dimensions (µm)		
Kernel Diameter	350	500
Buffer thickness	100	65
IPyC thickness	35	35
SiC thickness	35	35
OPyC thickness	40	40
Particle diameter	770	850
Material Densities (g/cm^3)		
Kernel	10.5	10.5
Buffer	1.0	1.0
IPyC ("sink/float" measurement)	1.87	1.87
SiC	3.2	3.2
OPyC (bulk density measurement)	1.83	1.83
Elemental Content Per Particle (µg)		
Carbon	305.7	379.9
Oxygen	25.7	61.6
Silicon	104.5	133.2
Uranium	254.1	610.2
Total particle mass (µg)	690.0	1184.9

Table 3-3. Fuel Compact Design Parameters

Parameter	Design Limit
Diameter, mm	12.45
Length, mm	49.3
Volume, cm^3	6.0
Shim particle composition	H-451 or TS-1240 graphite
Shim particle size	99 wt % < 1.19 mm; 95 wt % < 0.59 mm
Shim particle density (g/cm^3)	1.74
Binder type	Thermosetting resin
Filler	Petroleum derived graphite flour
Matrix density (g/cm^3)	0.8 to 1.2
Volume fraction occupied by fissile particles in an average compact	0.17
Volume fraction occupied by fertile particles in an average compact	0.03
Number of fissile particles in an average compact	4310
Number of fertile particles in an average compact	520

Table 3-4. Provisional Service Conditions for VHTR Fuel

Performance Parameters	**GT-MHR**	**VHTR**
Core outlet temperature	850	950 -1000
Core power density	5.8	5.8
Fuel element design	10-row block	10-row block
Core Residence Time (EFPD)	425	Determined by core design
Burnup - Fissile (% FIMA) Burnup - Fertile (% FIMA)	26 7	26 7
Maximum Fast Neutron Fluence (E>29 fJ), (n/m^2)	5×10^{25}	5×10^{25}
Maximum Fuel Temperature (^oC):		
- normal operation	1250	[1400][5]
- accident conditions	<1600	[<1800]

[5] Numerical values in [square brackets] are provisional values which are subject to revision as the plant design and safety analysis evolve.

Table 3-5. Coating Integrity Required for VHTR Fuel

Parameter	Commercial GT-MHR		VHTR	
	≥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)]	[$\leq 6.0 \times 10^{-4}$]	[$\leq 3.0 \times 10^{-4}$]	[$\leq 1.2 \times 10^{-3}$]

Table 3-6. Provisional Fission Metal Release Limits

Reactor Plant	Type	COT[6] (^oC)	Allowable Core Fractional Release			
			Cs-137		Ag-110m	
			"Expected"	"Design"	"Expected"	"Design"
MHTGR	Steam-cycle	700	7.0×10^{-6}	7.0×10^{-5}	5.0×10^{-4}	5.0×10^{-3}
HHT	Direct-cycle	850	2.0×10^{-5}	1.0×10^{-4}	8.6×10^{-5}	6.5×10^{-4}
GT-MHR	Direct-cycle	850	1.0×10^{-5}	1.0×10^{-4}	2.0×10^{-4}	2.0×10^{-3}
VHTR	Process heat	950	[1.0×10^{-5}]	[1.0×10^{-4}]	[2.0×10^{-4}]	[2.0×10^{-3}]

[6] COT = core outlet temperature

Table 3-7. Fuel Specification for UO_2^* Particle

Kernel		
Kernel Feature	Specification	
Composition	UO_2	
Diameter	350 μm	
Density	>10	
Oxygen-to-Metal Ratio	2.0	
ZrC Overcoating		
Coating Description	Coating Thickness	Coating Density (g/cc)
PyC Seal Coat	[5]	1.85 – 1.9
ZrC	[15]	[6.7]
TRISO Coating		
Buffer Layer	100	<1.0
IPyC	35	1.85 – 1.9
SiC	35	3.2
OPyC	40	1.85 – 1.9

Table 3-8. Fuel Specification for TRIZO Particle

Kernel		
Kernel Feature	Specification	
Composition	UCO	
Diameter	350 μm	
Density	>10	
O/U Ratio	[<0.15]	
TRISO Coating		
Coating Description	Coating Thickness	Coating Density (g/cc)
Buffer Layer	100	<1.0
IPyC	35	1.85 – 1.9
ZrC	35	[6.7]
OPyC	40	1.85 – 1.9

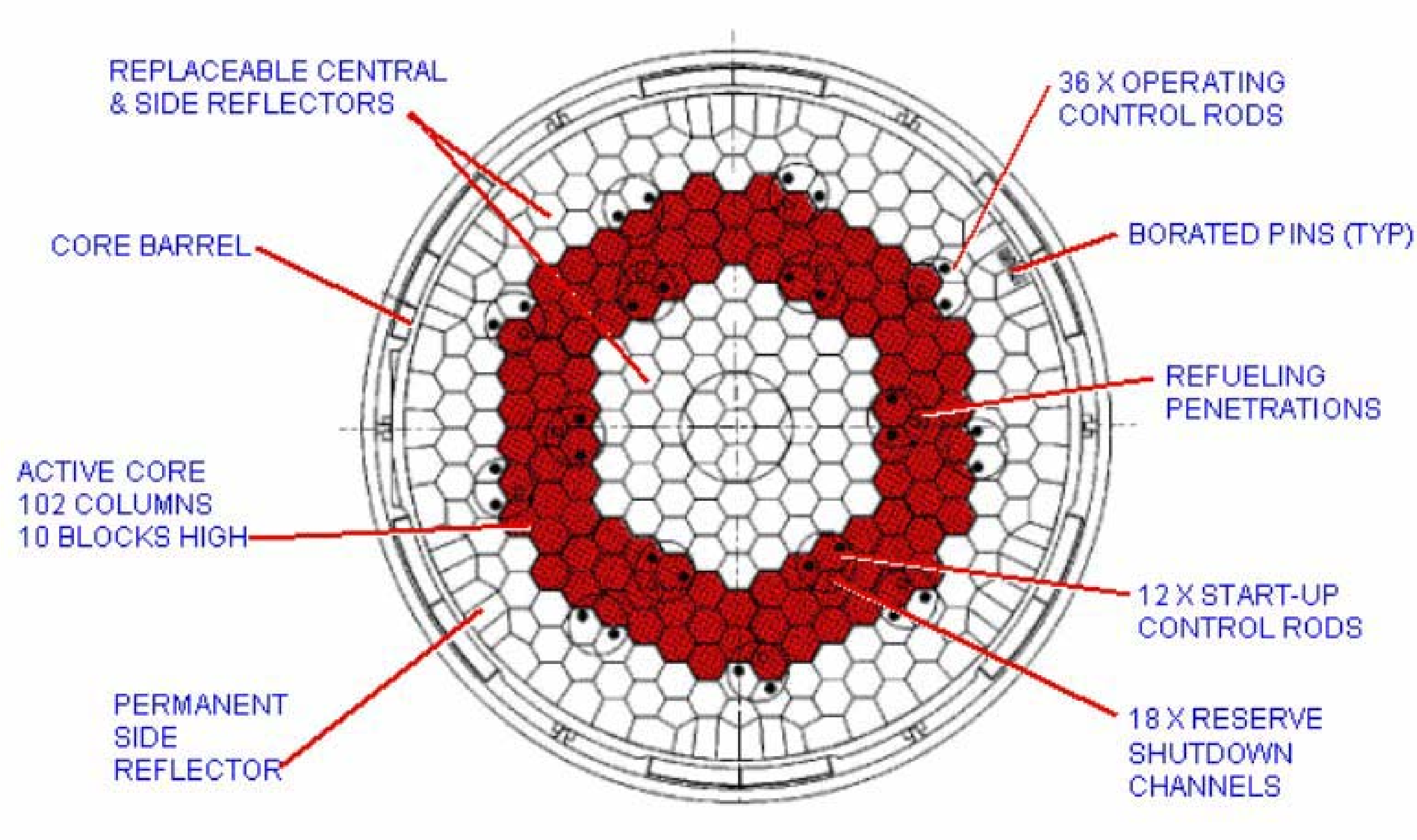


Figure 3-1. GT-MHR Core Layout

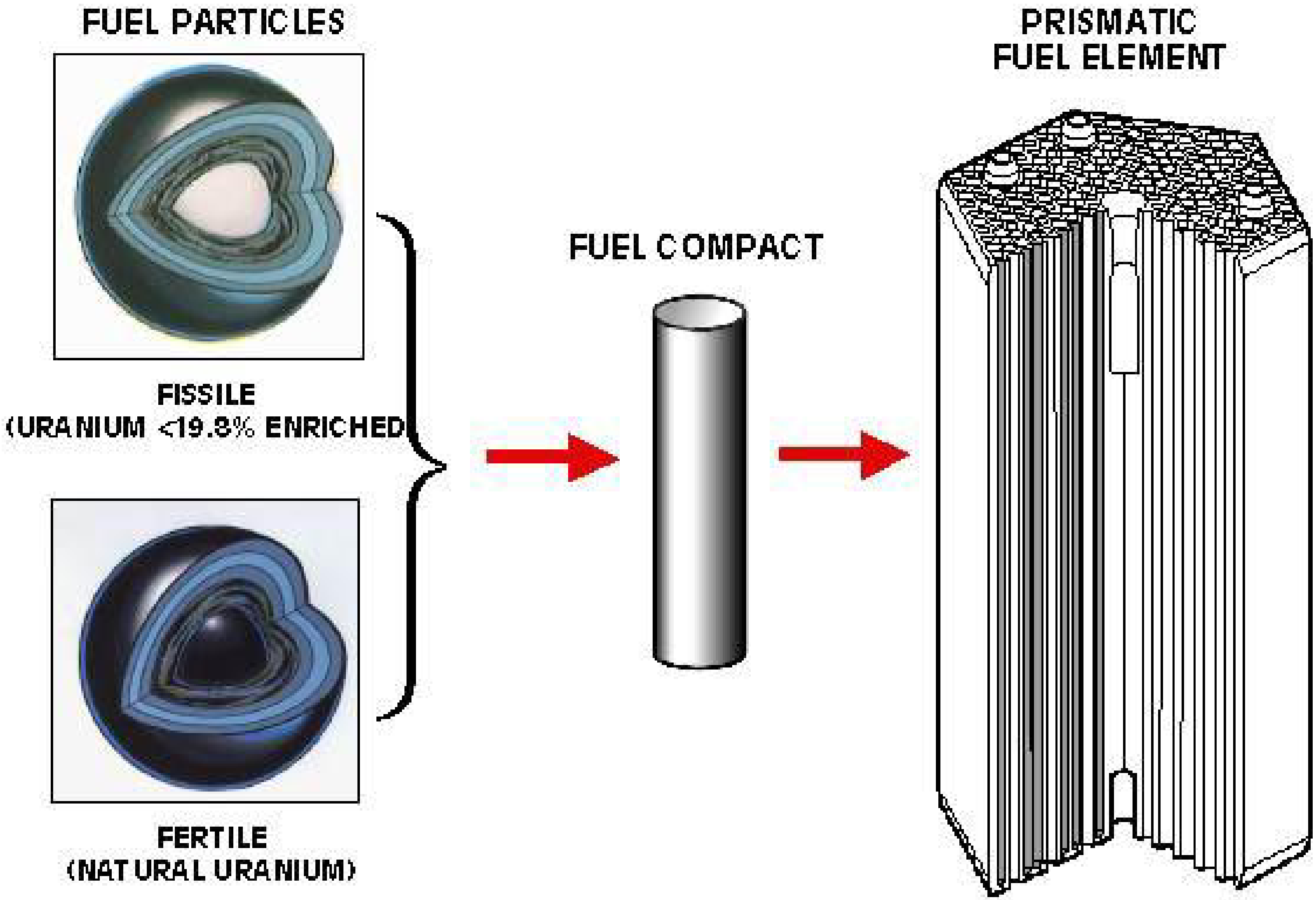


Figure 3-2. Fuel Element Components

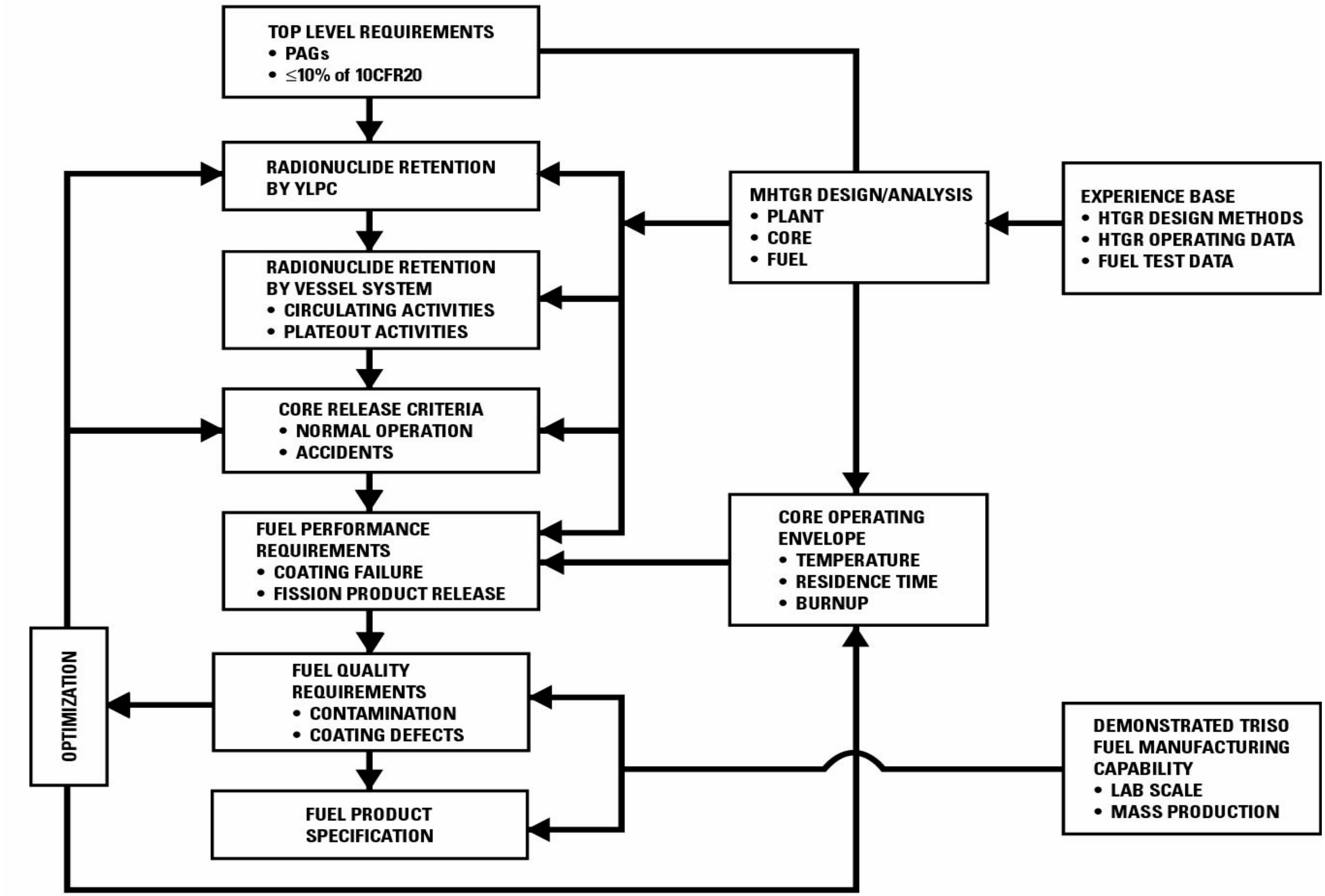


Figure 3-3. Logic for Deriving Fuel Quality Requirements

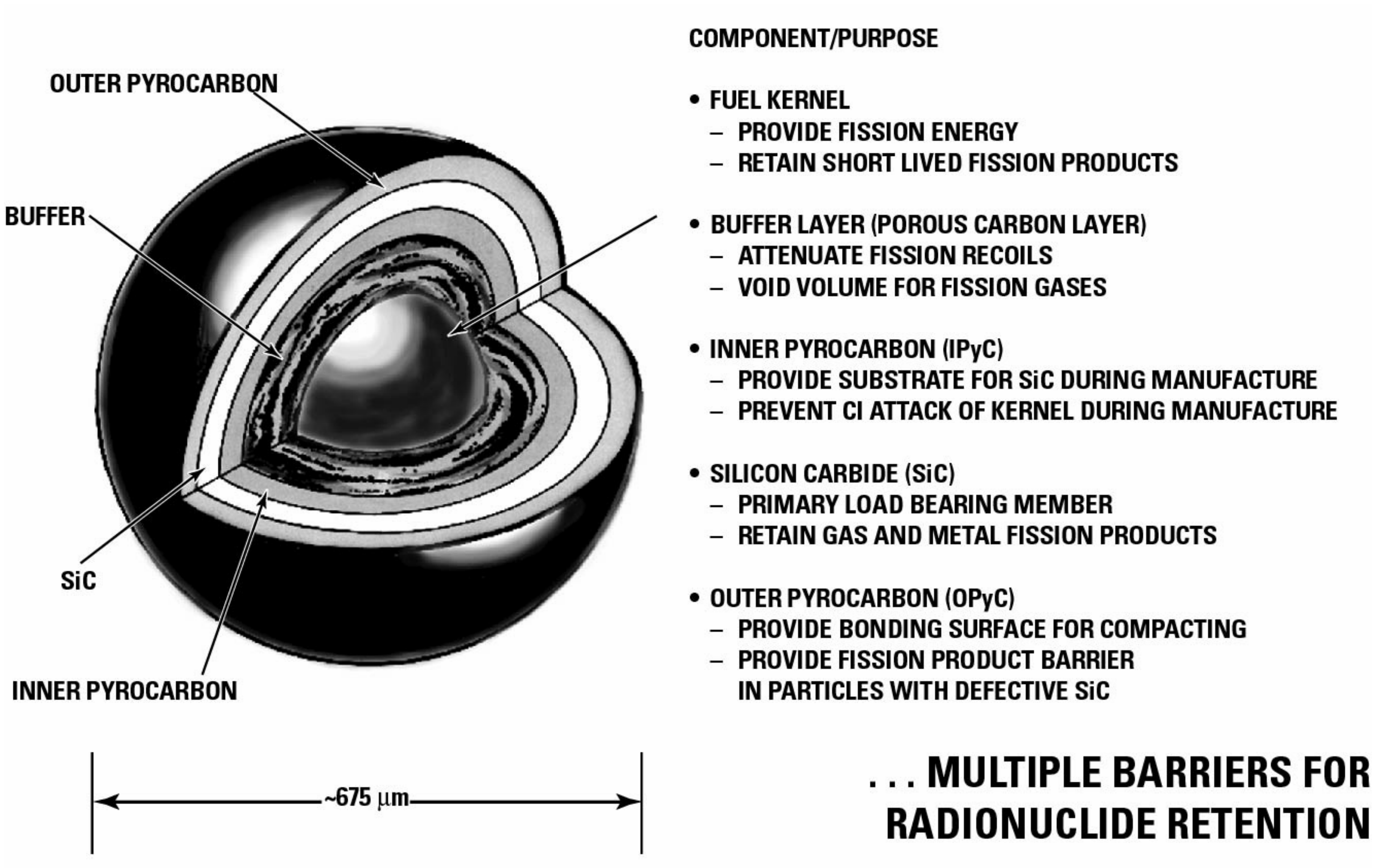


Figure 3-4. TRISO Coating System

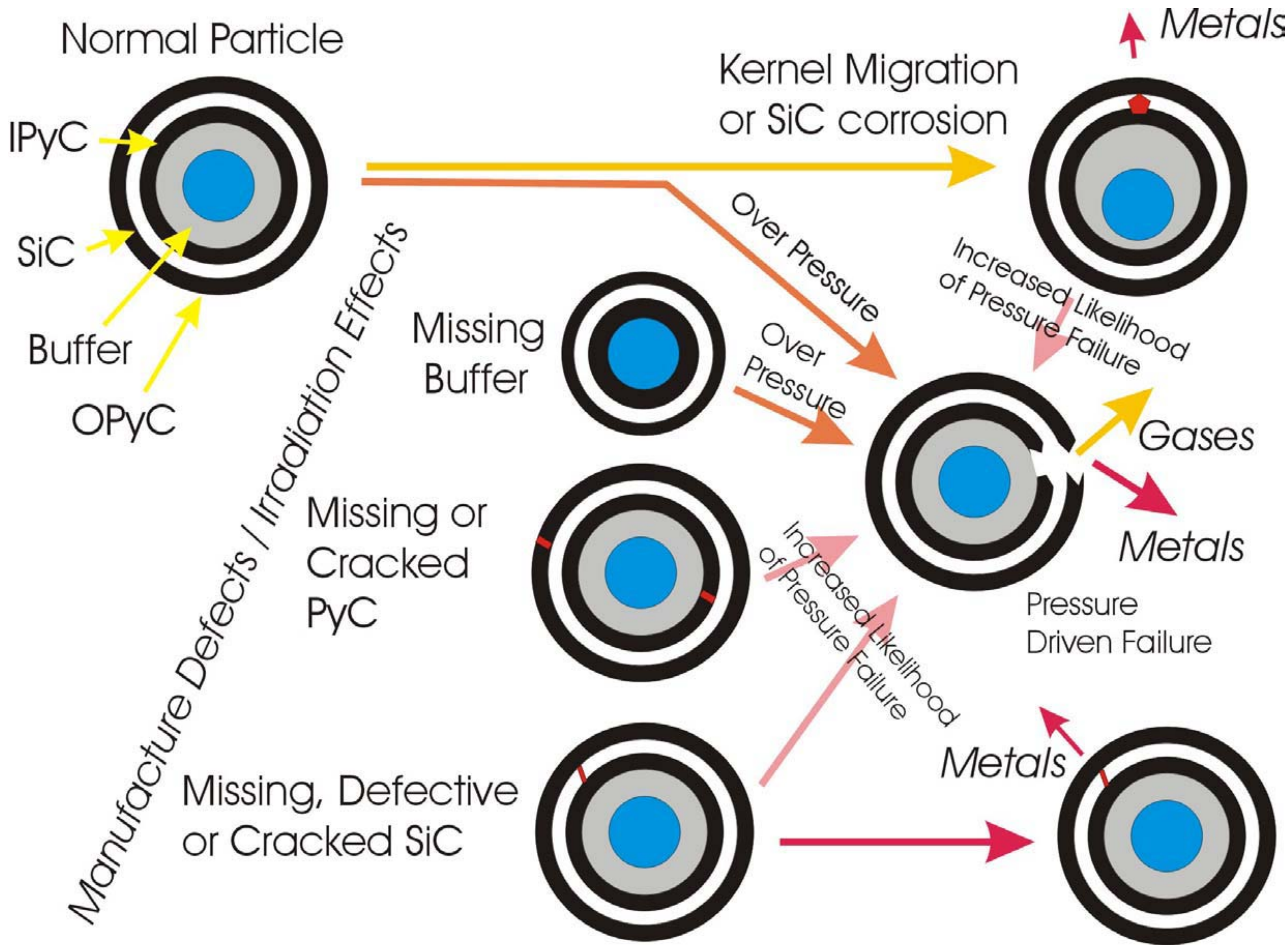


Figure 3-5. Particle Failure Mechanisms

4. Design Data Needs

The Design Data Needs related to fuel performance and fission product transport for the VHTR are summarized below.

4.1 Methodology for Defining Design Data Needs

As previously discussed, the five barriers for retaining radionuclides within the boundary of the reactor plant are: (1) the fuel kernel, (2) the particle coatings, (3) the fuel element graphite, (4) the primary coolant pressure boundary, and (5) the reactor containment building. 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 is to derive the allowable radionuclide release rates from the reactor 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.

The reactor designer must make certain assumptions about coated-particle fuel performance and radionuclide transport behavior, especially during the 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 DDNs as part of the functional analysis process (DDN Procedure 1986); the essence of this methodology is illustrated in Fig. 4-1.

4.2 Basis for VHTR Fuel/Fission Product DDNs

The source materials for developing the VHTR fuel/fission product DDNs were those fuel/fission product DDNs and development plans prepared by GA, INEEL 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 VHTR.

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

Qualitatively, the fuel/fission product DDNs for the VHTR can be summarized as follows:

1. Develop and qualify the fabrication processes needed to manufacture advanced coated-particle fuels with the attributes and as-manufactured quality required by the fuel product and process specifications.

2. Validate the fuel performance models that are used to predict fuel coating integrity for VHTR service conditions.[7]

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 fission product release from the core and transport in the primary coolant circuit during normal operation and accidents.

As previously introduced, it is assumed that this advanced fuel program is an incremental program with the DOE/NE-sponsored AGR fuel development program providing the base technology. In addition, the joint DOE-NNSA/MINATOM International GT-MHR program should contribute timely data to satisfy a number of generic fuel/fission product DDNs. Consequently, Table 4-1 indicates the anticipated programmatic sources of data to satisfy the various fuel/fission product DDNs: "AGR" refers to the AGR fuel program, "RF" refers to the International GT-MHR program, and "VHTR" refers to this program. The technology programs to satisfy the latter subset of DDNs are presented in Section 5.

4.3 References for Section 4

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.

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

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

[7] Oxidation of ZrC is more rapid than oxidation of SiC; therefore, special attention will be given to the characterization of ZrC oxidation kinetics and dependence on temperature and oxidant type and concentration (see DDNs VHTR 2.05 and 2.10).

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

Table 4-1. VHTR Fuel/Fission Product Design Data Needs

DDN No.	DDN Title	Data Source
Fuel Process Development		
VHTR.01.01	UCO Kernel Process Optimization	AGR
VHTR.01.02	UO_2* Kernel Process Development	**VHTR**
VHTR.01.03	TRISO Coating Process Optimization	AGR, RF
VHTR.01.04	ZrC Coating Process Development	**VHTR**
VHTR.01.05	Processes for Depositing Nonconventional Refractory Coatings	**VHTR**
VHTR.01.06	Fuel Compact Fabrication Process Development	AGR, RF
VHTR.01.07	Quality Control Test Techniques Development	AGR/**VHTR**
VHTR.01.08	Fuel Product Recovery Development	AGR
Fuel Materials Development		
VHTR.02.01	PyC/SiC Coating Material Property Data	AGR, RF
VHTR.02.02	ZrC Coating Material Property Data	**VHTR**
VHTR.02.03	Defective Particle Performance Data	AGR/**VHTR**
VHTR.02.04	Fuel Compact Thermophysical Properties	AGR, RF
VHTR.02.05	Thermochemical Performance Data for TRISO Fuel[8]	AGR, RF
VHTR.02.06	Irradiation Data for TRISO-coated UO_2* Particles	**VHTR**
VHTR.02.07	Irradiation Data for ZrC-coated Particles	**VHTR**
VHTR.02.08	Screening Data for Particles with Refractory Coatings (e.g., NbC, etc.)	**VHTR**
VHTR.02.09	Normal Operation Validation Data for Advanced Fuel	**VHTR**
VHTR.02.10	Accident Validation Data for Advanced Fuel[8]	**VHTR**
Fission Product Transport		
VHTR.03.01	Fission Gas Release from UCO Kernels	AGR
VHTR.03.02	Fission Gas Release from UO_2* Kernels	**VHTR**
VHTR.03.03	Fission Metal Diffusivities in UCO Kernels	AGR
VHTR.03.04	Fission Metal Diffusivities in UO_2* Kernels	**VHTR**
VHTR.03.05	Fission Metal Diffusivities in SiC Coatings	AGR, RF
VHTR.03.06	Fission Metal Diffusivities in ZrC Coatings	**VHTR**
VHTR.03.07	Screening Data for Metal Diffusivities in Refractory Coatings	**VHTR**
VHTR.03.08	Fission Product Diffusivities/Sorptivities in Graphite	AGR
VHTR.03.09	Tritium Permeation in Heat Exchanger Tubes	**VHTR**
VHTR.03.10	Tritium Transport in Core Materials	**VHTR**
VHTR.03.11	Radionuclide Deposition Characteristics for Structural Materials	AGR/**VHTR**
VHTR.03.12	Decontamination Protocols for Turbine Alloys	RF
VHTR.03.13	Radionuclide Reentrainment Characteristics for Dry Depressurization	AGR, RF
VHTR.03.14	Radionuclide Reentrainment Characteristics for Wet Depressurization	TBD
VHTR.03.15	Characterization of the Effects of Dust on Radionuclide Transport	AGR

[8] Includes studies of ZrC oxidation

DDN No.	DDN Title	Data Source
VHTR.03.16	Fission Product Transport in a Vented Low-Pressure Containment	AGR, RF
VHTR.03.17	Decontamination Efficiency of Depressurization Train Filter	AGR
VHTR.03.18	Fission Gas Release Validation Data for Advanced Fuel	**VHTR**
VHTR.03.19	Fission Metal Release Validation Data	**VHTR**
VHTR.03.20	Plateout Distribution Validation Data	AGR, RF
VHTR.03.21	Radionuclide “Liftoff” Validation Data	AGR, RF
VHTR.03.22	Radionuclide “Washoff” Validation Data	TBD

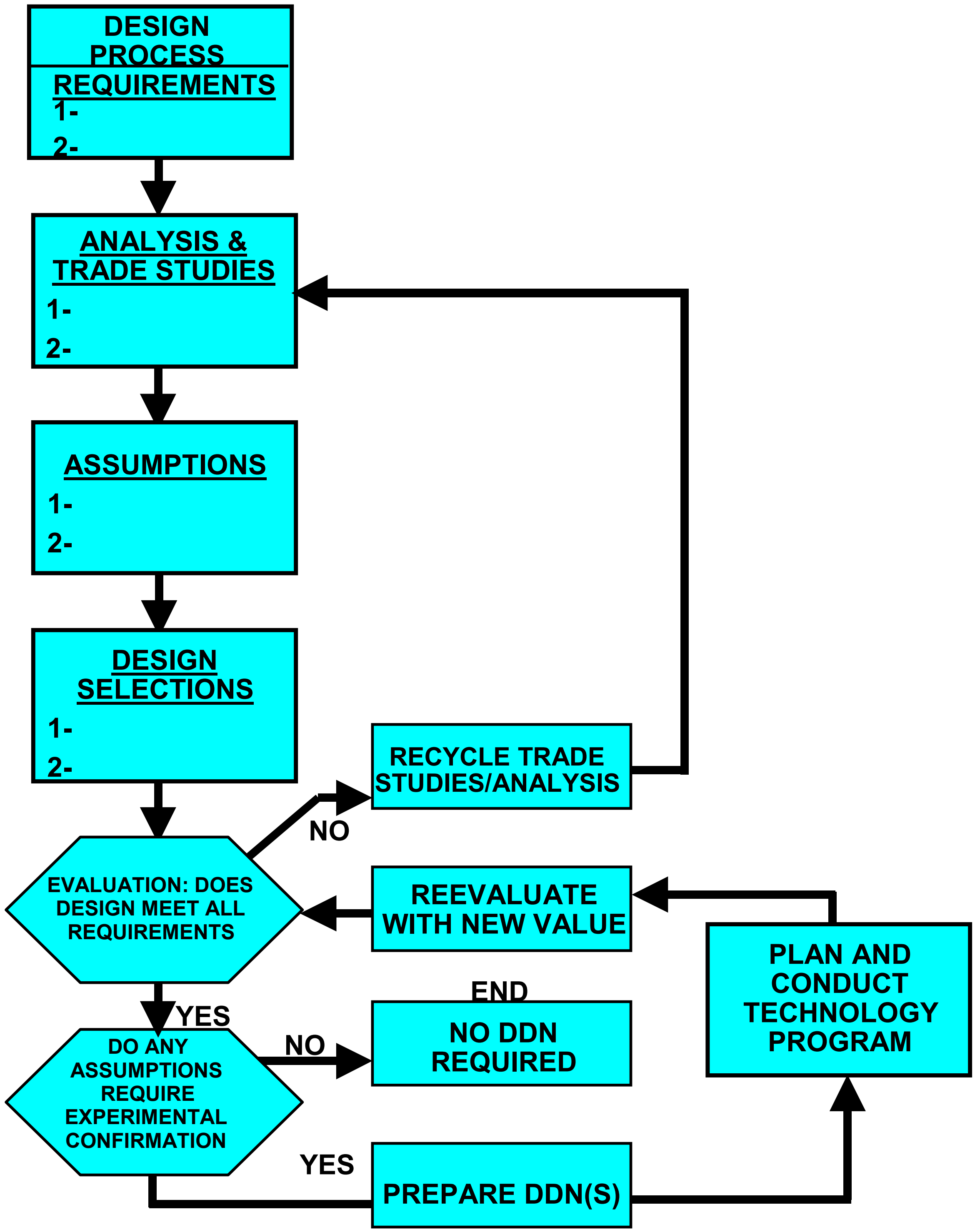


Figure 4-1. Process for Identifying DDNs

5. Technology Development Programs

The formal goal of this screening program is to contribute to the resolution of the fuel/fission product DDNs (Section 4) necessary to support the design, licensing, construction and operation of a VHTR Demonstration Module on a government site in full compliance with all applicable requirements. The logic and assumptions upon which this proposed program is predicated are summarized below.

1. The goal of the VHTR advanced fuel program is develop and qualify a fuel particle that that has performance capabilities beyond that of conventional TRISO (SiC) particles; such an advanced particle would meet performance requirements in a core characterized by sustained temperatures up to 1400 ^{o}C during normal operation and temperatures up to [2000 ^{o}C] during bounding core heatup accidents.
2. This program is an incremental program for which the AGR program provides the base technology; thus, it is assumed that the AGR program will fully characterize the irradiation behavior of TRISO-coated UCO particles up to 1400 ^{o}C and 26% FIMA and the post-irradiation heating behavior up to 2000 ^{o}C, including the effects of air and water ingress on TRISO particles.
3. Selection of a reference advanced fuel particle will be on the basis of irradiation capsules VHTR-1 and VHTR-2 and the subsequent post-irradiation heating of compacts from these two capsules at 1600 and 2000 ^{o}C.
4. Screening of advanced ("exotic") particle designs will be based upon irradiation capsule VHTR-6 and the subsequent post-irradiation heating of compacts from this capsule at 1800, 2000 and 2200 ^{o}C.
5. Most post-irradiation heating tests will be done primarily with high-burnup compacts, assuming that burnup effects up to 26% FIMA will be sufficiently small that this degree of conservatism can be tolerated without due penalty.
6. If accidents involving unlimited air ingress into the core or involving large water ingress are determined to be included in the licensing basis, then additional tests with reference fuel will be required to quantify the kinetics of particle failure and fission product release as a function of time, temperature and oxidant concentration.

A Work Breakdown Structure is presented in Table 5-1 to organize and manage these advanced fuel development activities. The Fuel Design tasks **(WBS 1.0)** define the design requirements and programmatic requirements which provide top-level goals and context for the Fuel Development tasks **(WBS 2.0)**.

The Fuel Process Development tasks **(WBS 2.1)** focus on the equipment and recipes (flowsheet, process conditions, procedures, etc.) required to fabricate the particles and compacts. The fuel process development tasks are described in the umbrella AF Plan.

Fuel Materials Development tasks **(WBS 2.2)** define the performance of the fuel under expected normal operating and accident conditions to confirm that the performance requirements imposed by the reactor designer can be met. The tasks includes measurements on test fuel performed during irradiation, post-irradiation examination, and accident condition testing. The Radionuclide Transport tasks **(WBS 2.3)** covers radionuclide transport in the reactor core and primary coolant circuit. A number of the Fuel Materials tasks and Radionuclide Transport tasks are closely coupled, and their assignment to a particular WBS category is largely a matter of convenience.

5.1 Fuel Materials Development

Fuel Materials Development **(WBS 2.2)** 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, and during post-irradiation examination.

5.1.1 Out-of-Pile Characterization

WBS 2.2.1: 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 (e.g., Lindemer 1976, Miller 1985, Gruebmeier 1977, and Goodin 1989). Thermochemical data exist for TRISO coated uranium and thorium-based fuel kernels, but no explicit database exists for UO_2^* and TRIZO particle designs. These various particle 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 UO_2^* and TRIZO particles under anticipated VHTR service conditions for normal operation and postulated accidents. These thermochemical studies will include: (1) basic studies to confirm oxygen management strategies, (2) studies to define potential attack of SiC and ZrC by fission products and CO, and (3) tests to confirm the materials properties as a function of neutron exposure. 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.

5.1.2 Irradiation Testing

WBS 2.2.2: An irradiation test program will be conducted to provide a basis for selecting and qualifying an advanced fuel that can meet projected VHTR performance requirements (Section 3.2). The irradiation program will have three phases: (1) screening of candidate fuels, (2) qualification testing of the reference fuel, including margin tests, and (3) validation testing of the reference fuel.

The AGR program is at this writing designing a new irradiation capsule with six independently operated and monitored cells, with each cell containing six fuel compacts,[9] for use in the Advanced Test Reactor (ATR) at INEEL (AGR Plan 2003). This six-cell capsule design will be complex and challenging; most facilities that have irradiated coated-particle fuels have used standard irradiation capsule designs with a single cell (e.g., HFIR at ORNL) or up to four independent cells (e.g., HFR Petten, R2 Studsvik, etc.). Consequently, following the precedent

[9] The cell configuration of the final capsule design is undetermined at this writing; i.e., it is not clear whether the six compacts would be in three fuel columns each containing two compacts or in two fuel columns each containing three compacts.

of DB-MHR fuel development plan (DB-MHR Plan 2002), it was conservatively assumed here that a four-cell capsule design, with each cell containing at least six fuel compacts (two fuel columns with three compacts per column per cell) will eventually be adopted as the standard irradiation capsule for the AGR program and for this program. If the AGR program succeeds in designing and qualifying a six-cell capsule, it will be used on this program as well, and the test articles and operating conditions of two the four cells defined herein will be replicated in the two additional cells to obtain better particle statistics and more on-line fission gas release data.

A series of fuel irradiation capsules will be required to satisfy the fuel/fission product DDNs identified in Section 4. A matrix of the planned capsule tests and the DDNs that they will address is given in Table 5-1. Summary descriptions of the capsules are given in Tables 5-2 (DDNs addressed), 5-3 (test objectives) and 5-4 (test conditions), and the description of each capsule type is elaborated in the following subsections.

The irradiation program will be coordinated with the process development effort to evaluate candidate particle 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 a reference advanced particle design. Qualification tests, including margin tests, will be conducted to define the sensitivity of this reference fuel design to variations in exposure conditions and to define its ultimate performance limits. The final test will be 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 (e.g., capability for detecting a single particle failure). Fuel performance statistical requirements are to be defined as determined 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 qualification tests, since these tests are used to screen candidates and identify boundaries and do not require as high a resolution.

5.1.2.1 Screening of Candidate Particle Designs

WBS 2.2.2.1: In two multi-cell capsules (**VHTR-1** and **VHTR-2**) there will be candidate UO_2^* designs (both UO_2^*-C and UO_2^*-B variants) and TRIZO particle designs. The purpose of these tests will be to compare performance of these candidate advanced particle designs and to select the design with the best high-temperature performance. After completion of the PIE and the post-irradiation heating tests on the irradiated particles, the data will be analyzed, and a selection of reference advanced particle design will be made. Follow-on capsules will irradiate the reference particle design. Design optimization will be based on the results of ongoing thermochemical, structural, and nuclear analysis as well experimental results.

VHTR-6 is a screening irradiation of several advanced particle designs (e.g., “exotic” refractory coatings such as TaC fission product getters in the kernel, etc.) which promise superior irradiation and accident performance at very high temperatures. Irradiation and accident performance data will be generated to determine if more “exotic” particle designs promise sufficiently superior performance at high temperature and burnup and/or under oxidizing conditions to merit further development. It will be conducted later in the program (irradiation beginning in FY2013, see Section 8) after the qualification tests for the reference fuel (see next subsection) have been completed irradiation. In the unexpected event that neither UO_2^* or

TRIZO fuel performed sufficiently well in the early capsule irradiations, this screening irradiation test of more "exotic" particle designs would likely be performed earlier.

5.1.2.2 Qualification Testing of Reference Fuel

WBS 2.2.2.2: Irradiation test **VHTR-3** is a test of reference fuel to peak VHTR service conditions (e.g., 1400 ^{o}C) to determine the effects of irradiation temperature on coating performance and to generate samples for post-irradiation heating tests with sufficient quantities to demonstrate compliance with VHTR requirements. It will also provide feedback for process optimization.

Irradiation test **VHTR-4** is a margin test that will focus on demonstrating the performance of reference fuel compacts under the most severe VHTR core combination of temperature, burnup, and fast fluence and beyond. The range of service conditions will be defined to determine those conditions which produce excessive fuel failure due to structural failure mechanisms (e.g., due to fluence effects) and thermochemical mechanisms. 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 **VHTR-1** and **VHTR-2** irradiations and the core analyses that will have been performed. Traditionally, a margin test would include service conditions (e.g., fast fluence, burnup 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" by causing sufficient coating failure that the failure rates and failure mode(s) can be reliably determined by conventional PIE techniques.

Irradiation test **VHTR-5** is a test of reference fuel, fabricated with optimized process conditions, to peak VHTR service conditions. The test conditions will be similar to those for **VHTR-3** since the **VHTR-5** test is essentially repeats the **VHTR-3** test with optimized fuel. It will provide additional quantities of irradiated reference fuel for post-irradiation heating tests which will include the effects of air and water ingress. It will also provide feedback to finalize product and process specifications.

5.1.2.3 Validation Testing of Reference Fuel

WBS 2.2.2.3: Irradiation test **VHTR-7** with reference fuel compacts, fabricated with an optimized design and using optimized processes, will be the final planned irradiation of reference fuel in the program. This validation test 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 **VHTR-7** capsule will expose fuel to conditions simulating VHTR core average temperature (~1000 °C), and core peak temperature (~1400 °C). The peak burnup will be ~29 % FIMA and the peak fast fluence will be 5.5 x 10^{25} n/m^2 which are about 10% beyond the expected VHTR design burnup and fast neutron fluence. 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).

(The remaining irradiation tests shown in Tables 5-2 through 5-4 are fission product transport tests which are described in Section 5.2.)

5.1.3 Post-irradiation Examination

WBS 2.2.3: The standard scope of the post-irradiation examination for each of the irradiation capsules described in Section 5.1.2 is presented in this section. These PIE workscopes and the task descriptions are nearly identical to those presented in the AGR Plan. A capsule PIE is composed of a number of tasks. Some of these tasks may be conducted in parallel while others must be conducted sequentially; for example, a capsule must be opened before any work can be done with the fuel so it is a serial task . Fuel compact deconsolidation can be a parallel task because only a portion of the compacts is used for the task, and the remainder of the compacts can used for other unrelated tasks. The actual sequencing of the tasks will be detailed in the PIE plan; but for planning purposes, it may be assumed that a PIE will take approximately one year to complete with no restrictions on resources. The following tasks outline the options that are likely to be available for a particular PIE.

The scope of each PIE would be reevaluated based upon the on-line fission gas release data obtained during the irradiation; if these gas release data indicated unexpected behavior in a particular cell, that cell would likely undergo additional examination during PIE in attempt to identify the cause(s) of the unexpected behavior. The tasks that are planned for the individual capsules are shown in Table 5-5.

As a point of departure for developing the PIE plans, the following PIE tasks will be conducted, as appropriate, for the irradiation capsules.

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

PIE TASK-2: Capsule Gamma-Scanning: Prepare the capsule for gamma scanning, and gamma scan the capsule. Produce a color-coded map of the capsule (based upon the local gamma emission intensity) and any regions that appear abnormal.

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

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

PIE TASK-5: Fuel Compact Cross-Section: Examine cross sections of a fuel compact by optical metallography to document conditions within the compact, including fuel particles and matrix. The examination will visually document conditions within fuel particles (e.g., kernel migration, kernel morphology, buffer integrity, integrity of the individual structural layers, chemical attack of the individual layers, etc.).

PIE TASK-6: Fuel Compact R/B and Reactivation: Place fuel compacts, one at a time, in a TRIGA or TRIGA-like reactor with an internal temperature-controlled furnace. This task will allow R/B measurements of individual fuel compacts (rather than the total R/B from a fuel body containing several fuel compacts) and the identification of compacts with damaged fuel particles.

It will also regenerate measurable inventories of short-lived radionuclides, including radiologically significant 8-day I –131 so that their release characteristics can be measured during subsequent post-irradiation heating tests.

PIE TASK-7: Component Activity: Individually gamma-count capsule components to determine the isotopes and amount of fission products present. It may be necessary to leach certain components and count the leach solutions.

PIE TASK-8: Leach-Burn-Leach: Measure coating failure fractions in selected irradiated fuel compacts using the leach-burn-leach technique.

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

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

PIE TASK-11: Fuel Metallography: Examine both intact and failed fuel particles to document failure mechanism in the coatings using optical metallography (the interface between the IPyC and the ZrC coating in TRIZO particles and the ZrC overcoat in UO_2^* particles are of particular interest).

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

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

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

PIE TASK-15: Properties of Irradiated Materials Specimens: Measure properties (thermal, physical, mechanical) on samples of irradiated materials, such as kernels and coatings.

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

PIE TASK-17: Fission Product Release During Post-irradiation Heating: Conduct post-irradiation heating tests to measure fission product release as a function of time at temperatures in the range of 1400 – 2200 °C. These safety tests can be performed on fuel compacts or loose fuel particles. The test facility must accommodate three atmospheric compositions for these heating tests: pure helium, helium/air, and helium/steam.

PIE TASK-18: Postheating Metallography: Characterize coating layer integrity by optical metallography to identify and quantify coating failure mechanisms. Evidence of layer thinning and/or decomposition, chemical attack and the mechanical state and microstructures of the layers are of particular interest.

PIE TASK-19: Postheating SEM: Measure (map) fission product distribution (especially Pd, Ag, and Cs) in fuel particles (kernels, buffer, coating layers) and fuel compacts (i.e., in the compact matrix) with an SEM/microprobe (WDX) identify and quantify coating failure mechanisms. Evidence of fission product accumulations at the coating interfaces, fission product attack of the coatings, and fission products outside the fuel particles is of particular interest.

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

PIE TASK-21: Reporting: Disseminate the findings, results, and problems of the PIE task in both formal and informal reporting. Costs and schedules for each capsule are provided in Section 8.

5.1.4 Post-irradiation Heating Tests

WBS 2.2.4: A critically important part of the screening program presented in this plan is a series of post-irradiation heating tests to characterize the performance capabilities of advanced fuel particles under simulated core heatup accidents. These post-irradiation heating tests are an integral part of the post-irradiation examination program; however, they are of sufficient importance to merit a more complete description of the tests and their objectives.

5.1.4.1 Test Facility Construction

WBS 2.2.4.1:. The Core Conduction Cooldown Test Facility (CCCTF) is currently available at ORNL, and a new post-irradiation heating facility is planned to be constructed on the AGR program, perhaps at Argonne National Laboratory – West (ANL-W). Additional post-irradiation heating facilities will have to be constructed to support this program since it would be conducted concurrently with the AGR program.

According to the AGR Fuel Plan, the new AGR heating facility will have "…the capability to work with air and steam ingress conditions at the temperatures of programmatic interest." The heating facilities needed for this program must permit heating irradiated fuel compacts and loose particles in dry helium to 2200 ^{o}C and heating in helium/air and helium/steams mixtures to at least 1400 ^{o}C. Whether the new AGR design will accommodate those test conditions is uncertain at this writing. If it does, then the facility design can simply be replicated for use on this program with an attendant cost savings; if not, then additional design work will be need to be funded by this program.

The AGR program also plans to develop and commission a new facility for performing fission gas release (release rate-to-birth rate, R/B) measurements on irradiated fuel compacts and loose particles and for reactivation of irradiated fuel compacts and particles prior heating to produce measurable inventories of short-lived radionuclides, including radiologically significant 8-day I-131. This plan assumes that AGR program will accomplish this goal in a timely fashion and that this facility will be able for use on this program as well.

5.1.4.2 Test Matrix

The planned post-irradiation heating tests are summarized in Table 5-6, and the test objectives are elaborated in the following subsections. Unless otherwise stated, all of the tests are "ramp/hold" tests as performed by the Germans (e.g., IAEA TECDOC-978 1997) where the test fuel is heated at 50 ^{o}C/min to the desired temperature and maintained at a constant temperature for the duration of the test. As an option, periodic holds at intermediate temperatures can be introduced, but this complicates data interpretation and correlation. In the final validation tests, a variable time-temperature history approximating a core heatup accident will be used.

WBS 2.2.4.2/2.2.4.3: Two irradiated UO_2^*-C compacts and two irradiated UO_2^*-B compacts recovered from capsule **VHTR–1** and two irradiated TRIZO compacts from **VHTR–2** will be heated at 1600 and 2000 ^{o}C in dry helium for up to 500 hr or until significant coating failure is evident from the periodic fission product release measurements. The 1600 ^{o}C temperature was chosen for continuity with the existing international post-irradiation heating data base for conventional TRISO-coated particles, and the 2000 ^{o}C temperature was chosen as reasonable performance capability desired for an advanced particle design. This set of heating data (**PIH-1** through **PIH-6**) along with on-line fission gas release measurements and other PIE data for capsules **VHTR–1** and **VHTR–2** will be the primary technical basis for choosing a reference advanced particle design for further qualification testing.

WBS 2.2.4.4: Six reference fuel compacts from capsule **VHTR–3** will be heated for up to 500 hr or until significant coating failure is evident from the fission product release measurements. Heating tests **PIH-7** through **PIH-9**, along with the earlier heating data from **VHTR–1** or **VHTR–2** (depending upon which fuel type is chosen as the reference fuel) will begin to determine the effects of irradiation exposure (burnup and fast fluence), if any, on high temperature fuel performance. Comparison of the **PIH-8** data at 2000 ^{o}C to the earlier data at 2000 ^{o}C from **VHTR–1** or **VHTR–2** will determine if process optimization has had any effect on performance. Heating tests **PIH-10** through **PIH-12** will begin to determine the effects of air and steam on particle performance. These tests could be critically important, especially if TRIZO is chosen as the reference particle, since ZrC is less oxidation resistant than SiC.

WBS 2.2.4.5: Two reference fuel compacts from capsule **VHTR–4** will be heated for up to 500 hr or until significant coating failure is evident from the periodic fission product release measurements. Capsule **VHTR–4** is a margin test of reference fuel which presumably will have taken the fuel to well beyond VHTR service conditions to induce 0.1 – 1.0% coating failure (i.e., to find the irradiation performance limits in fast fluence and temperature); consequently, much of the fuel may be uncharacteristically degraded at the end of the irradiation. Heating test **PIH-13** at 2000 ^{o}C will use an irradiated fuel compact that experienced severe irradiation conditions for comparison with the other 2000 ^{o}C heating data to determine the effects on accident performance. Heating test **PIH-14** at 2200 ^{o}C will use an irradiated fuel compact that experienced less severe irradiation conditions (cell 4) to determine the effects of high post-irradiation temperatures on performance.

WBS 2.2.4.6: Six compacts of optimized reference fuel from capsule **VHTR–5** will be heated for up to 500 hr or until significant coating failure is evident from the periodic fission product release measurements. Heating tests **PIH-15** through **PIH-17**, along with the earlier heating data from **VHTR–3,** will determine the effects of irradiation exposure (burnup and fast fluence), if any, on high temperature fuel performance. Heating tests **PIH-18** through **PIH-20,** along with the earlier heating data from **VHTR–3**, will determine the effects of air and steam on particle performance. If the optimized reference fuel in capsule **VHTR–5** performs fundamentally different (better) than the earlier reference fuel in **VHTR–3**, then additional heating tests may be necessary with the optimized fuel to obtain higher confidence in the performance data.

WBS 2.2.4.7: Four fuel compacts from capsule **VHTR–6** will be heated for up to 500 hr or until significant coating failure is evident from the periodic fission product release measurements. Capsule **VHTR–6** is planned to be a test of at least two advanced fuel designs with greater high-temperature, high-exposure potential and greater oxidation resistance than the reference fuel (additional variants may be included as loose-particle piggyback samples). The irradiation **VHTR–6** of will have taken the fuel to well beyond VHTR service conditions. Heating tests **PIH-21** and **PIH-22** will heat the two leading variants (in fuel compacts) at 2200 ^{o}C for comparison with the reference fuel at 2200 ^{o}C. The better performing variant will be tested further in heating tests **PIH-23** and **PIH-24** to determine its oxidation resistance at 1400 and 1800 ^{o}C (the former temperature for comparison with the data for the reference fuel).

WBS 2.2.4.8: Four compacts from capsule **VHTR–7,** containing reference fuel made to the final process specifications, will be heated with variable time-temperature history approximating a core heatup accident; the peak temperatures in tests **PIH-25** and **PIH-28** will range from 1400 to 2000 ^{o}C. Capsule **VHTR–7** and its associated heating tests are intended to provide integral test data for validating the design methods developed from the early test data. Depending upon design and licensing considerations, one or two of the lower temperature tests may include air ingress.

(The remaining post-irradiation heating tests shown in Table 5-6 are fission product transport tests which are described in Section 5.2.)

5.2 Radionuclide Transport

The fission product transport work scope **(WBS 2.3)** in support of VHTR design and licensing is planned to be consistent with the overall program goal of providing validated radionuclide transport methods by the year 2015. DDNs VHTR 03.01 - 03.22 (Table 4-1) provide definition of the required data. The primary objective is to determine if advanced fuels, specifically UO_2^* and TRIZO fuels, present any significantly new radionuclide release behavior beyond that already observed with current TRISO-coated fuels.

The work scope addressed in this screening plan **(WBS 2.3.1)** is limited to radionuclide transport in kernels, particle coatings, and fuel-compact matrix. As indicated in Table 4-1, radionuclide transport in the fuel-element graphite, primary coolant circuit and in the reactor building are largely generic topics which are to be addressed by the AGR program. There are exceptions, e.g., tritium transport in core materials (DDN VHTR.03.10), which are not currently addressed in the AGR Plan; hence, they are addressed in the umbrella development plan.

5.2.1 Radionuclide Transport in Reactor Core

As introduced in Section 3, 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 plan, the test article collectively representing these three release barriers is referred to a "fuel body." It consists of fuel 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 transport characteristics in kernels, coatings, and matrix/graphite, and (2) integral release data from the entire assembly representing radionuclide 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.

5.2.1.1 RN Transport during Normal Operation

WBS 2.3.1.1.1: Capsule **VHTR-8** will be designed to characterize the fission product release failed UO_2^* and TRIZO particles.[10] The kernels and finished particles will be taken from the same production runs that produced the test fuel for capsules **VHTR-1** and **VHTR-2.** To provide a known fission product source, laser failed (LF) or "designed-to-fail" (DTF)[11] particles will be seeded in the compacts. The quantity of LF or DTF particles will be approximately 10 times the expected number of normally failed particles so that the fission product source is quantifiable and the releases are measurable; this fraction is expected to be in the 0.1–1.0% range. Based upon previous investigations, the releases from LF or DTF particles are judged to be comparable to releases from actual failed particles

The test articles in the four cells of capsule **VHTR-8** are defined in Table 5-4. At this writing, the plan is to use seeded UO_2^*-ZrC-overcoat/TRISO compacts in two cells, seeded UO_2*-ZrC-buffer/TRISO compacts in a single cell, and seeded TRIZO compacts in a single cell. The rationale for this cell allocation is that the kernel release characteristics of the UO_2^*-ZrC-buffer/TRISO particle should be similar to the kernel release characteristics of a standard UO_2 kernel which has been well characterized by the Germans, and the release characteristics of the UCO kernel in the TRIZO particle should be similar to that of the UCO kernel in a standard TRISO particle which has been partially characterized and will be further characterized by the AGR program.

[10]As for the Fuel Materials irradiations (Section 5.1), the planning basis here was a 4-cell capsule. The AGR program is planning to utilize a multi-cell capsule design with six independently purged and operated cells. If this design proves viable, it will be used here as well.

[11] "Designed-to-fail" (DTF) particles, as used here, are standard 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.

The underlying assumption is that the ZrC in UO_2^* serves primarily as a physical barrier to fission product release; however, this assumption is debatable. The ZrC will clearly getter excess oxygen, thereby lowering the oxidation potential in the kernel, which may change the chemical speciation of volatile fission products which, in turn, may affect their transport behavior (and should suppress kernel migration in UO_2 kernels). These chemical potential considerations also argue for using laser-failed rather than "designed-to-fail" particles in this test. If a six-cell design is qualified, it would be prudent to devote two cells to each of the three fuel variants. This issue will be revisited with the test specification for this capsule is prepared and the AGR program capsule design is finalized.

Loose failed and intact particles of each of the three fuel variants would also be included in sealed piggyback samples in each cell for subsequent fission product release experiments, e.g., to characterize fission metal diffusivities in the ZrC coatings of intact TRIZO particles.

In order to determine the temperature dependence of fission gas release, capsule **VHTR-8** will be thermally cycled to the extent practical by varying the composition of capsule purge gas (e.g., from pure helium to purge neon); it is anticipated that this will allow a 100-150 ^{o}C variation in fuel temperature from the design temperature specified in Table 5-4. This thermal cycling will be done periodically over the full range of burnup.

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.

WBS 2.3.1.1.2: Tentatively, the post-irradiation examination of capsule **VHTR-8** will include the tasks indicated in Table 5-5. These plans will be reviewed when the PIE specification is prepared. 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.

In addition to the PIE tasks called out in Table 5-4, the piggyback samples irradiated in capsule **VHTR-8** will be recovered and characterized. A number of them will undergo specialized examinations and post-irradiation heating to characterize radionuclide transport rates in the kernels and coatings; the Ag diffusivity in the ZrC coating of the TRIZO particle is of particular interest because of the paucity of data. (These tests will be elaborated in the umbrella plan because the emphasis is on the screening tests.)

WBS 2.3.1.1.7: In addition to the differential data to be obtained from capsule **VHTR-8** 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 VHTR core. Capsule **VHTR-9** is designed to provide these integral validation data.

The test article in capsule **VHTR-9** will be compacts with optimized reference fuel from the same production run as validation capsule **VHTR-7** which have been seeded with missing-buffer (MB) reference particles. 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. Capsule service conditions are given in Table 5-4. In contrast to **VHTR-8**, capsule **VHTR-9** 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 capsule design and service conditions will be carefully reviewed when the test specification is prepared. 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. Selected fuel compacts and/or whole fuel bodies will be reserved for accident testing.

WBS 2.3.1.1.8: Tentatively, the post-irradiation examination of capsule **VHTR-9** will include the tasks indicated in Table 5-5. These plans will be reviewed when the PIE specification is prepared.

5.2.1.2 RN Transport under Accident Conditions

The new post-irradiation heating facility, similar to the CCCTF but with extended capabilities, will be able to heat irradiated loose particles, fuel compacts, and complete fuel bodies to 2200 ^oC in dry helium and up to at least 1400 ^oC in helium/air or helium/steam environments. The furnace purge gas will provide control of the atmosphere and means for continuous monitoring of fission gas release. There will be provisions for removable and replaceable deposition surfaces (essentially a "cold finger") to monitor fission metal release during the test. Posttest examinations will provide similar data as in the irradiation tests: individual particle failure fractions and fission metal release fractions from the IMGA measurements, metallographic examinations of kernels and coatings, and micro-scale examination by 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 prior to the PIH tests.

WBS 2.3.1.2.1: Irradiated compacts recovered from capsule **VHTR-8** will be reactivated prior to heating in a new test facility. In heating tests **PIH-29** through **PIH-31**, the three fuel variants in the capsule will be subjected to a series of temperature ramp/hold "steps" with fission product release measurements made at 1200, 1400, 1600 and 1800 ^oC; the duration of each of hold periods will depend upon observed release rates of the noble gases, especially 10.7-yr Kr-85; the primary goal is to measure the I-131 fractional release at each temperature step. If this proves impractical, then additional single-step ramp/hold tests may be necessary. Heating **PIH-32** test will be performed with the best performing variant of the candidate advanced fuels (based upon the then available data) at 1400 ^oC in an air/helium environment.

As stated previously, it has been assumed that the kernel release characteristics of failed UO_2^* particles will be similar to that of failed TRISO-coated UO_2 particles which have been well characterized by the Germans and that the release characteristics of failed TRIZO particles will be similar to that of failed TRISO-coated UCO particles which have been partially characterized (no I-131 release data are currently available) and will be further characterized by the AGR program. The four compact heating tests proposed are intended to confirm this assumption. If the assumption proves invalid, then additional heating tests, including cheaper tests with failed loose particles recovered from piggyback samples, will be necessary to develop reliable fission gas release correlations.

WBS 2.3.1.2.6: Entire irradiated fuel bodies, or sections thereof with the fuel compact(s) encased in graphite (e.g., a wedge containing one fuel stack), from **VHTR-9** will be subjected to accident condition testing to obtain integral release data for design methods validation. In heating tests **PIH-33** through **PIH-35**, the reactivated fuel specimens will be subjected to a variable time-temperature history representative of that experienced in the reactor core; it is anticipated that a peak transient temperature of 1800°C will be specified for the tests. One or more of the tests may include air, depending upon the results of the previous heating data and the results of the detailed safety analysis for the VHTR Demonstration Module that will have been completed by that time. The detailed test matrix will be finalized when the test specification is prepared

5.3 References for Section 5

[AGR Plan] "Technical Program Plan for the Advanced Gas Reactor Fuel Development and Qualification Program" ORNL/TM-2002/262, Oak Ridge National Laboratory, April 2003.

[DB-MHR Plan] "Deep-Burn Modular Helium Reactor Fuel Development Plan," GA-224-0-TRT-000167, General Atomics, September 2002.

Goodin, D. T., "US/FRG Accident Condition Fuel Performance Models," DOE-HTGR-85107, Rev. 1, General Atomics, 1989.

Gruebmeier, H., et al., "Silicon Carbide Corrosion in High-Temperature Gas-Cooled Reactor Fuel Particles," *Nucl Tech.* **35**, pp. 413, 1977.

Lindemer, T. B., R. L. Pearson, "Kernel Migration for HTGR Fuels from the Th-U-C-O-N System," TM-5207, ORNL, April 1976.

Lindemer, T. B., "Thermochemical Analysis of Gas-Cooled Reactor Fuels Containing Am and Pu Oxides," ORNL/TM-2002/133 (September 2002).

McCardell, R. K., et al., "NP-MHTGR Fuel Development Program Plan," EGG-NPR-8971, Rev. C., EGG Idaho, Inc., September 1992.

Table 5-1. Work Breakdown Structure for Advanced Fuel Development

1. Fuel Design
    - 1.1 Design Data Needs
    - 1.2 Fuel Development Plan
    - 1.3 Fuel Specifications
    - 1.4 Model Development
        - 1.4.1 Particle Performance
        - 1.4.2 Radionuclide Transport
    - 1.5 Design Methods Validation
2. Fuel Development
    - 2.1 Fuel Process Development
        - 2.1.1 Kernel Process Development
            - 2.1.1.1 UCO Kernel Optimization
            - 2.1.1.2 UO_2* Kernel Development
            - 2.1.1.3 Advanced Kernel Process Development
        - 2.1.2 Coating Development
            - 2.1.2.1 TRISO Coating Process Optimization
            - 2.1.2.2 ZrC Coating Process Development
            - 2.1.2.3 Processes for Nonconventional Coatings
        - 2.1.3 Compact Development
        - 2.1.4 Quality Control Test Techniques Development
        - 2.1.5 Fuel Product Recovery Development
    - 2.2 Fuel Materials Development
        - 2.2.1 Out-of-Pile Characterization
        - 2.2.2 Irradiation Testing
            - 2.2.2.1 Screening Tests
            - 2.2.2.2 Qualification Tests
            - 2.2.2.3 Validation Tests
        - 2.2.3 Post-irradiation Examination
        - 2.2.4 Accident Simulation Testing
    - 2.3 Radionuclide Transport
        - 2.3.1 Transport in Reactor Core
            - 2.3.1.1 Normal Operation
            - 2.3.1.2 Accident Conditions
        - 2.3.2 Transport in Primary Coolant Circuit
            - 2.3.2.1 Normal Operation
            - 2.3.2.2 Accident Conditions

Table 5-2. Irradiation Tests to Satisfy VHTR DDNs

Capsule	Description[12]	Design Data Need											
		02.02	02.06	02.07	02.08	02.09	02.10	03.02	03.04	03.06	03.07	03.18	03.19
VHTR-1	Screening – UO_2^*		X										
VHTR-2	Screening – TRIZO	X		X						X			
VHTR-3	Qualification – Ref. Fuel	X	X	X									
VHTR-4	Margin – Ref. Fuel	X	X	X									
VHTR-5	Qualification – Ref. Fuel	X	X	X									
VHTR-6	Screening – Adv. Particles				X						X		
VHTR-7	Validation – Ref. Fuel					X	X						
VHTR-8	FP Transport - Particles		X	X				X	X	X			
VHTR-9	Validation – FP Release											X	X

[12] Test matrix assumes that either UO_2^* or TRIZO (ZrC-coated UCO) will be selected as "Reference" fuel after screening capsule irradiations are completed

Table 5-3. Planned Irradiation Tests to Develop Advanced VHTR Fuel

Capsule	Description	Primary Objective/Expected Result
VHTR-1	<u>Screening – UO_2^*</u> Irradiation of UO_2^*-Zr buffer and UO_2^*-ZrC overcoat to peak VHTR service conditions	Irradiation data for the two UO_2^* designs to determine if they meet VHTR fuel requirements, and generation of samples for post-irradiation heating tests. Provide feedback for process optimization. Sufficient performance data to permit the selection of the reference VHTR fuel particle for further qualification testing. Provide feedback for process optimization.
VHTR-2	<u>Screening – TRIZO</u> Irradiation of particles with UCO kernels and a ZrC coating replacing the SiC layer ("TRIZO")	Irradiation data for TRIZO particles to determine if they meet VHTR fuel requirements, and generation of samples for post-irradiation heating tests. Provide feedback for process optimization. Sufficient performance data to permit the selection of the reference VHTR fuel particle for further qualification testing. Provide feedback for process optimization.
VHTR-3	<u>Qualification – Reference Fuel</u> Irradiation of reference VHTR fuel (assumed either UO_2^* or TRIZO) in statistically significant quantities	Irradiation data for reference VHTR fuel and generation of samples for post-irradiation heating tests with sufficient quantities to demonstrate compliance with VHTR performance requirements. Provide feedback for process optimization.
VHTR-4	<u>Margin – Reference Fuel</u> Irradiation of reference VHTR fuel in statistically significant quantities to sufficiently high fast fluences and temperatures to cause 0.1 – 1.0% coating failure	Determine ultimate performance limits of reference VHTR fuel under irradiation and simulated accident conditions, including massive air ingress. Provide feedback for process optimization.
VHTR-5	<u>Qualification – Reference Fuel</u> Irradiation of reference VHTR fuel (assumed either UO_2^* or TRIZO) fabricated with optimized process specifications in statistically significant quantities	Irradiation data for reference VHTR fuel fabricated with optimized process specifications and generation of samples for post-irradiation heating tests with sufficient quantities to demonstrate compliance with VHTR performance requirements. Provide feedback to finalize product and process specifications.
VHTR-6	<u>Screening – Advanced Particles</u> Irradiation of several advanced	Sufficient irradiation and accident performance data to determine if more

Capsule	Description	Primary Objective/Expected Result
	particle designs (e.g., "exotic" coatings, getters, etc.) which promise superior irradiation and accident performance at very high temperatures	"exotic" particle designs (e.g., refractory coatings such as TaC, kernel getters, etc.) promise sufficiently superior performance at high temperature and/or under oxidizing conditions to merit further development.
VHTR-7	Validation – Reference Fuel Irradiation of reference VHTR fuel fabricated to final product and process specifications in full-size coater to peak VHTR service conditions under near real-time test conditions	Irradiation data for reference VHTR fuel and generation of samples for post-irradiation heating tests with sufficient quantities to provide experimental basis for validating the design methods for predicting fuel performance.
VHTR-8	FP Transport – Particles Irradiation of "designed-to-fail" UO_2^*-ZrC overcoat, UO_2^*-ZrC buffer, and TRIZO particles seeded in fuel compacts and of loose failed and intact UO_2^*-C, and UO_2^*-B, and TRIZO particles in piggy-back samples.	Characterization of the fission product release rates from failed and intact UO_2^*-ZrC overcoat, UO_2^*-ZrC buffer, and TRIZO particles under irradiation and core heatup conditions. Early data will contribute to selection of reference fuel for VHTR.
VHTR-9	Validation – FP Release Irradiation of fuel compacts seeded with missing-buffer reference particles under peak VHTR core conditions	Irradiation data for reference VHTR fuel and generation of samples for post-irradiation heating tests with sufficient quantities to provide experimental basis for validating the design methods for predicting fission product release from the VHTR core.

Table 5-4. Capsule Design Conditions

Capsule	Cell[13]	Test Article Kernel/Coating	Configuration	Fast Fluence (n/m^2)	Temperature (^{O}C)	Fissile Burnup (% FIMA)	Piggyback Samples
VHTR-1	1	UO_2^*- Zr buffer TRISO	Compacts Loose particles	<5.0 x 10^{25}	1000	<26	Yes
	2A/2B	UO_2^*- Zr buffer TRISO	Compacts	5.0 x 10^{25}	1350	26	No
	3A/3B	UO_2^*- ZrC overcoat TRISO	Compacts	5.0 x 10^{25}	1350	26	No
	4	UO_2^*- ZrC overcoat TRISO	Compacts Loose particles	<5.0 x 10^{25}	1000	<26	Yes
VHTR-2	1A/1B	TRIZO UCO/ZrC	Compacts Loose particles	<5.0 x 10^{25}	1000	<26	Yes
	2A/2B	TRIZO UCO/ZrC	Compacts	5.0 x 10^{25}	1350	26	No
	3	TRIZO UCO/ZrC	Compacts	5.0 x 10^{25}	1350	26	No
	4	TRIZO UCO/ZrC	Compacts Loose particles	<5.0 x 10^{25}	1000	<26	Yes
VHTR-3	1	Reference Fuel (UO_2^* or TRIZO)	Compacts	<5.0 x 10^{25}	800	<26	Yes
	2A/2B	Reference Fuel (UO_2* or TRIZO)	Compacts	5.0 x 10^{25}	1200	26	Yes

13 "XA/XB" indicate replicated cells in the event that a capsule design with six independent cells is used rather than a four-cell capsule design as assumed here.

Capsule	Cell[13]	Test Article Kernel/Coating	Configuration	Fast Fluence (n/m^2)	Temperature (^{O}C)	Fissile Burnup (% FIMA)	Piggyback Samples
	3A/3B	Reference Fuel (UO_2* or TRIZO)	Compacts	5.0×10^{25}	1400	26	No
	4	Reference Fuel (UO_2* or TRIZO)	Compacts	$<5.0 \times 10^{25}$	1000	<26	Yes
VHTR-4	1	Reference Fuel (UO_2* or TRIZO)	Compacts	$<5.0 \times 10^{25}$	>1400	<26	Yes
	2A/2B	Reference Fuel (UO_2* or TRIZO)	Compacts	$>>5.0 \times 10^{25}$	1350	>26	Yes
	3A/3B	Reference Fuel (UO_2* or TRIZO)	Compacts	$>5.0 \times 10^{25}$	>1400	>26	No
	4	Reference Fuel (UO_2* or TRIZO)	Compacts	5.0×10^{25}	1250	26	Yes
VHTR-5	1	Reference Fuel (Optimized process)	Compacts	$<5.0 \times 10^{25}$	800	<26	No
	2	Reference Fuel (Optimized process)	Compacts	5.0×10^{25}	1200	26	No
	3A/3B	Reference Fuel (Optimized process)	Compacts	5.0×10^{25}	1400	26	No
	4A/4B	Reference Fuel (Optimized process)	Compacts	$<5.0 \times 10^{25}$	1000	<26	No
VHTR-6	1	Advanced Particles (Variant #1)	Compacts Loose particles	5.0×10^{25}	1350	26	Yes (Variant #3)

Capsule	Cell[13]	Test Article Kernel/Coating	Configuration	Fast Fluence (n/m^2)	Temperature (^{O}C)	Fissile Burnup (% FIMA)	Piggyback Samples
	2A/2B	Advanced Particles (Variant #1)	Compacts Loose particles	>5.0 x 10^{25}	1500	>26	Yes (Variant #3)
	3A/3B	Advanced Particles (Variant #2)	Compacts Loose particles	>5.0 x 10^{25}	1500	>26	Yes (Variant #4)
	4	Advanced Particles (Variant #2)	Compacts Loose particles	5.0 x 10^{25}	1350	26	Yes (Variant #4)
VHTR-7	1	Reference Fuel (Final specs)	Compacts	<5.0 x 10^{25}	900	<26	No
	2A/2B	Reference Fuel (Final specs)	Compacts	5.5 x 10^{25}	1200	≥29	No
	3A/3B	Reference Fuel (Final specs)	Compacts	5.5 x 10^{25}	1400	≥29	No
	4	Reference Fuel (Final specs)	Compacts	<5.0 x 10^{25}	1100	<26	No
VHTR-8[14]	1	Failed Particles UO_2*-C/TRISO	Compacts Loose particles	<5.0 x 10^{25}	1000[15]	<26	Yes
	2A/2B	Failed Particles UO_2^{*}-C/TRISO	Compacts Loose particles	5.0 x 10^{25}	1400	26	Yes

[14] For the two fission product transport capsules - VHTR-8 and VHTR-9 - it may be desirable to operate the two additional cells at different temperatures in the event that a capsule design with six independent cells is used rather than a four-cell capsule design as assumed here; the final operating conditions will be defined in the test specifications.

[15] In contrast to the Fuel Materials capsules, the temperatures in the VHTR-8 fission product release capsule will be thermally cycled to the extent feasible by varying the composition of the sweep gas in order to determine the temperature dependence of the fission gas release.

Capsule	Cell[13]	Test Article Kernel/Coating	Configuration	Fast Fluence (n/m^2)	Temperature (^{O}C)	Fissile Burnup (% FIMA)	Piggyback Samples
	3A/3B	Failed Particles UCO/ZrC	Compacts Loose particles	5.0×10^{25}	1200	26	Yes
	4	Failed Particles UO_2^*-B/TRISO	Compacts Loose particles	$<5.0 \times 10^{25}$	1200	<26	Yes
VHTR-9[8]	1	Reference Fuel (MB particles)	Seeded Compacts	$<5.0 \times 10^{25}$	800[16]	<26	No
	2A/2B	Reference Fuel (MB particles)	Seeded Compacts	5.0×10^{25}	1200	26	No
	3A/3B	Reference Fuel (MB particles)	Seeded Compacts	5.0×10^{25}	1400	26	No
	4	Reference Fuel (MB particles)	Seeded Compacts	$<5.0 \times 10^{25}$	1000	<26	No

[16] The given values represent the maximum temperature; the cells will be operated with a variable time-temperature approximating the reactor core.

Table 5-5. Capsule PIE Matrix

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

Table 5-6. Post-irradiation Compact Heating Test Matrix

Test	Test Article (Burnup)*	Temperature (^{O}C)	Time History	Test Objectives[17]						
				1	2	3	4	5	6	7
PIH-1	VHTR-1 (High)	1600	Ramp/hold	X						
PIH-2	VHTR-1 (High)	1600	Ramp/hold	X						
PIH-3	VHTR-1 (High)	2000	Ramp/hold	X						
PIH-4	VHTR-1 (High)	2000	Ramp/hold	X						
PIH-5	VHTR-2 (High)	1600	Ramp/hold	X						
PIH-6	VHTR-2 (High)	2000	Ramp/hold	X						
PIH-7	VHTR-3 (High)	1800	Ramp/hold		X					
PIH-8	VHTR-3 (High)	2000	Ramp/hold		X					
PIH-9	VHTR-3 (Low)	2000	Ramp/hold		X					
PIH-10	VHTR-3 (High)	1600	Ramp/hold				X			
PIH-11	VHTR-3 (High)	1600	Ramp/hold				X			
PIH-12	VHTR-3 (High)	1600	Ramp/hold					X		

[17]Objective 1. Performance Margin Test
Objective 2. Irradiation Exposure Effects
Objective 3. Irradiation Temperature Effects
Objective 4. Air Ingress Effects
Objective 5. Water Ingress Effects
Objective 6. Fission Product Release
Objective 7. Methods Validation Test

*"(High)" denotes High burnup
"(Low)" denotes Lower burnup

Test	Test Article (Burnup)*	Temperature (^{O}C)	Time History	Test Objectives[17]						
				1	2	3	4	5	6	7
PIH-13	VHTR-4 (High)	2000	Ramp/hold							
PIH-14	VHTR-4 (High)	2200	Ramp/hold							
PIH-15	VHTR-5 (High)	2000	Ramp/hold	X						
PIH-16	VHTR-5 (Low)	1800	Ramp/hold		X	X				
PIH-17	VHTR-5 (Low)	1800	Ramp/hold			X				
PIH-18	VHTR-5 (High)	1400	Ramp/hold				X			
PIH-19	VHTR-5 (High)	1400	Ramp/hold				X			
PIH-20	VHTR-5 (High)	1400	Ramp/hold					X		
PIH-21	VHTR-6 (High)	2200	Ramp/hold	X						
PIH-22	VHTR-6 (High)	2200	Ramp/hold	X						
PIH-23	VHTR-6 (High)	1400	Ramp/hold	X			X			
PIH-24	VHTR-6 (High)	1800	Ramp/hold	X			X			
PIH-25	VHTR-7 (High)	1400	Core simulation							X
PIH-26	VHTR-7 (High)	1600	Core simulation							X
PIH-27	VHTR-7 (High)	1800	Core simulation							X
PIH-28	VHTR-7 (High)	2000	Core simulation							X
PIH-29	VHTR-8 (High)	1800	Step/hold[18]						X	

[18]"Step/hold" indicates a series of temperature ramp/hold "steps" with RN release measurements at 1200, 1400, 1600 and 1800 ^{o}C.

Test	Test Article (Burnup)*	Temperature (^{O}C)	Time History	Test Objectives[17]						
				1	2	3	4	5	6	7
PIH-30	VHTR-8 (High)	1800	Step/hold						X	
PIH-31	VHTR-8 (High)	1800	Step/hold						X	
PIH-32	VHTR-8 (High)	1400	Ramp/hold				X		X	
PIH-33	VHTR-9 (High)	1400	Core simulation						X	X
PIH-34	VHTR-9 (High)	1600	Core simulation						X	X
PIH-35	VHTR-9 (High)	1800	Core simulation						X	X

6. Facility Requirements for VHTR Fuel Development Plan

6.1 Test Facility Requirements

This program has demands for specialized facilities and equipment. Comprehensive reviews of the existing US and international facilities currently available to perform such coated-particle fuel R&D are provided in both the 2003 AGR fuel plan and the 2002 DB-MHR fuel plan; portions of each are excerpted below. As previously stated, this advanced fuel plan is structured to be an incremental plan with the AGR program providing the base technology. It is no surprise then that the facility requirements for this program largely match those for the AGR program. The DB-MHR program has similar facility requirements, especially regarding irradiation facilities; but it has additional highly specialized requirements, especially in the fuel process development area, because of the necessity of handling and processing significant quantities of plutonium and higher actinides.

The facility requirements are summarized in the subsequent sections along with brief discussions of existing facilities and capabilities that can be used to fulfill the requirements. The AGR will require essentially the same facilities; consequently, the two programs will need to be carefully coordinated. If the requisite facilities will not be available when required or do not exist (e.g., new post-irradiation heating facilities), they have been provided for in the cost estimates for this program.

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

1. Fuel fabrication process development (detailed in umbrella AF Plan)

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 the capability of existing facilities at ORNL and INEEL and how they might be used in the program.

6.2 Description of Test Facilities

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

<u>Fuel fabrication and characterization at ORNL</u>:

- Metals and Ceramics Research and Development Laboratory, Building 4508
- Process Development Laboratory, Building 4501

- Post Irradiation Examination Laboratory, Building 4501

Irradiation Testing:

- High Flux Isotope Reactor, Building 7910 (ORNL)
- Advanced Test Reactor (INEEL)
- High Flux Reactor- Petten (The Netherlands)
- SM-3 & RBT-6 (Research Institute of Atomic Reactors (RIAR), Dimitrovgrad, RF)

Post Irradiation Examination:

- Post Irradiation Examination Facility, Building 3525 (ORNL)
- ATR hot cells (INEEL)
- ANL-W hot cells (Idaho)
- RIAR hot cells (Dimitrovgrad, RF)

Post-irradiation Heating Tests

- Core Conduction Cooldown Test Facility (ORNL)

6.2.1 Facilities for the Fabrication of Coated Particle Fuel

The once extensive facilities at GA in San Diego to produce coated-particle fuel have been completely dismantled. In addition, ORNL essentially dismantled equipment for fabricating fuel in the early 1980s, with the demise of its fuel cycle programs. However, many of the facilities capable of accommodating the fabrication process development 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, especially for fuel compacting. The capability to produce UO_2 and UCO kernels and to produce coated particles still exists at BWX Technologies (BWXT; formerly B&W) in Lynchburg.

The AGR program has already begun the process of recommissioning kernel and coating facilities at ORNL and BWXT. The fuel process facilities and specialized equipment needed for fuel fabrication will be elaborated in the umbrella AF Plan.

6.2.2 Facilities for Irradiation Testing of Fuel

The **High Flux Isotope Reactor** (HFIR) at ORNL is a light-water cooled, beryllium-reflected reactor that uses HEU U-Al fuel to produce high neutron fluxes for materials testing and isotope production. It has been used extensively in the U.S. gas reactor programs to irradiate coated-particle fuel. Two specific materials irradiation facilities are of note here. The large RB positions (of which there are eight) are 46 mm in diameter and 500 mm long and can accommodate capsules holding up to 24 compacts, (3 in each graphite body, 8 bodies axially) in a single purged cell. This configuration was used for the HRB-21 experiment, the last irradiation in the U.S. commercial program in the early 1990s. The small VXF positions (of which there are 16) are 40 mm in diameter and 500 mm long. They can accommodate capsules holding up to 16 compacts (8 in each graphite body, 2 bodies axially) in a single purged cell. This configuration was used for the NPR-1 and NPR-2 irradiations, the last two irradiations at ORNL under the NP-

MHTGR program in the early 1990s. There is a large axial flux gradient that must be considered in the design of any experiment in any of these locations. The building complex housing HFIR is depicted in Fig. 6-1; a view of the reactor and its storage basin is shown in Fig. 6-2.

The **Advanced Test Reactor** (ATR) at INEEL is a light-water-cooled, beryllium-reflected reactor that uses HEU U-Al fuel in a four-leaf clover configuration to produce high neutron fluxes for materials testing and isotope production. The clover leaf configuration results in nine very high flux positions, termed flux traps. In addition, numerous other holes of varying size are available for testing. Of interest here are several holes that can be used to irradiate coated-particle fuel. The large B holes in ATR (of which there are four) are 38 mm in diameter and 760 mm in length. They can accommodate five individually purged cells, with two graphite bodies per cell, containing up to three compacts per body. Thus, a total of 30 compacts can be irradiated in this location. Of special note here is the very flat burnup and fluence profiles available axially in the ATR over the 760-mm length. This allows for nearly identical irradiation of large quantities of fuel. The ATR was used extensively during the NP-MHTGR program to irradiate Li targets (ATR-1, ATR-2, ATR-3, and ATR-4 series of experiments) and fuel (NPR-1A irradiation) in the early 1990s. The building complex housing ATR is depicted in Fig. 6-3; a view of the reactor core and pool is shown in Fig. 6-4.

The **High Flux Reactor** ("HFR Petten") in Petten, the Netherlands, is a multipurpose research reactor that has many irradiation locations for materials testing. It has been the workhorse for irradiation of spherical fuel elements for the German HTR project in the 1970–1995 time frame. It has also irradiated GA compacts for the U.S. program in the late 1980s. They have two different types of irradiation rigs/locations in the facility: one that can accommodate compacts, and one that can accommodate spheres. The REFA and BEST rigs are multi-cell capsules, 63 to 72 mm in diameter, that can handle four to five spheres in up to four separate cells. The TRIO or QUATTRO rigs/locations are ~32 mm in diameter and 600 mm in useful length. They can handle three or four parallel stacks of compacts. For the three-stack configuration, about 30 compacts could, in principle, be irradiated in the rig. These rigs are currently dedicated to the EU-1 (sphere) and EU-2 (compact) irradiations under the HTR-F program in Europe. The current configurations of EU-1 and EU-2 are limited in the number of individually purged cells that are being used. In EU-2, only two cells are planned, one for German spheres and one for Chinese spheres. In EU-2, only one swept cell is planned for the U.S. compacts. In addition, there is a large axial flux gradient across the useable length (40% spread maximum to minimum) that must be considered in the design of any experiment.

Facilities for irradiation of coated-particle fuel are being established at the Russian Research Institute of Atomic Reactors (RIAR), Dimitrovgrad, RF, as part of the DOE/MINATOM International GT-MHR program. The use of two RIAR reactors is planned, the **SM-3** reactor and the **RBT-6** reactor. These reactors provide a variety of test channels and operating environments. The SM-3 reactor has higher neutron flux locations and can be used for testing of statistically significant numbers of particles in compacts and to produce irradiated compacts for accident testing. The lower flux RBT-6 can be used to test fuel compacts and loose particle samples and fuel material samples to obtain specific fuel material irradiation characteristics, fission product transport information, and produce irradiated material for special tests. Full burnup and full fast neutron fluence can be reached in a short time in the inner positions of SM-3. It is prudent, however, to restrict the heat generation in a particle and the rate of accumulation of burnup and fast fluence to less than a factor of three more than the GT-MHR; these limitations imply that full exposure can be accomplished in periods between about 300 days and 750 days in SM-3.

Coated-particle fuel irradiation capsules can be fitted into test "channels" in these reactors. Each apparatus is made up of “ampoules” (cells). Four channels in SM-3 are suitable for irradiation testing of coated particles. The irradiation capsule currently being designed for the GT-MHR program consists of three ampoules; each of the ampoules can accommodate four compacts; consequently, a maximum of 12 compacts can be tested in each channel and a maximum of 48 compacts can be tested simultaneously in the four SM-3 channels. In the RBT-6 there are eight suitable channels, and the RBT-6 irradiation apparatus has one ampoule which can accept one compact or a collection of loose particles. To reach full burnup and full fast fluence simultaneously, it is necessary to reduce the thermal flux by using neutron shields of materials such as hafnium.

6.2.3 Post Irradiation Examination and Test Facilities

The ORNL Post-Irradiation Examination Laboratory (Building 3525) is presently equipped to carry out the various functions associated with post-irradiation capsule disassembly and the subsequent examination of capsule components, fuel compacts, and fuel particles. PIE facilities at ORNL along with its status are summarized in Table 6-1. 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. These operations include disassembly, sectioning, radiography, metallography, dimensional measurements, and waste handling. Hot cell facilities are also available at INEEL and ANL-W, and these facilities have extensive experience in performing PIEs of nuclear fuels. They have less experience, hence less specialized equipment and expertise, with performing PIEs on coated-particle fuels.

The AGR program plans to develop or reestablish several PIE measurement capabilities. A new particle gas analyzer (PGA) to crush a particle at a specified temperature and analyze the released gases, including CO and CO_2, must be designed and constructed; a throughput of a least several particles per day is required. Specialized tools and techniques will to be developed to investigate the physical properties of irradiated coatings, especially the structure and anisotropy pyrocarbon coatings.

6.2.4 Post-Irradiation Accident Test Facilities

The Core Conduction Cooldown Test Facility (CCCTF) at ORNL is an existing furnace located in a hot cell which is specifically designed for heating irradiated coated-particle fuel compacts. The facility, shown schematically in Fig. 6-5, is designed for continuous monitoring of noble gas release during heating, and it has removable cold-finger for periodic determination of the fractional releases of condensable radionuclides, including radioiodines and volatile fission metals. The AGR program has plans to upgrade the existing CCCTF to allow testing with helium/air and helium/steam atmospheres and to replicate the upgraded facility, perhaps at INEEL or ANL-W. The AGR program will need two PIH facilities to perform the planned heating tests on the proposed schedule.

Some of the irradiated fuel compacts will need to be reactivated prior to heating in order to produce measurable quantities of radiologically important radionuclides, such as 8-day I-131.

One possibility under consideration by the AGR program for accomplishing this reactivation is to install a high-temperature King furnace in the TRIGA reactor at ANL-W. Such a facility, which would reestablish a capability that previously existed at GA, would permit not only reactivation irradiated fuel particles and compacts but also would permit R/B measurements on as-manufactured and irradiated fuel specimens.

6.3 Selected Test Facilities for Advanced Fuel Development

As previously stated, this plan is structured to be an incremental program with the AGR program providing the base technology; thus, facility requirements for this program largely match those for the AGR program. Consequently, this program will generally use the same test facilities as available or will replicate them when necessary and practical.

6.3.1 Facilities for the Fabrication of Advanced Particle Fuel

This program will use the fuel process development equipment and support services, such as QC laboratories, etc., as available at ORNL and BWXT. Certain equipment, such as a laboratory-scale coater, may be replicated and dedicated to this program (e.g., for ZrC coating development). Details will be elaborated in the umbrella AF Plan.

6.3.2 Facilities for Irradiation Testing of Advanced Fuel

As described in Section 5, this program for advanced fuel development requires multiple irradiations using a multi-cell capsule design. The ATR at INEEL is the irradiation facility of choice for this screening program. The AGR program is at this writing designing a new irradiation capsule with six independently operated and monitored cells, with each cell containing six fuel compacts, for use in the ATR. This six-cell capsule design will be complex and challenging; consequently, it was conservatively assumed here that a four-cell capsule design, with each cell containing at least six fuel compacts (two fuel columns with three compacts per column per cell) will eventually be adopted as the standard irradiation capsule for the AGR program and for this program. If the AGR program succeeds in designing and qualifying a six-cell capsule, it will be used on this program as well.

The HFIR was not chosen because only single-cell capsule designs are available for use there, and a single-cell capsule is ill suited for a screening program such as this one. On the other hand, if it were desirable to accelerate this program or compress the irradiation phase (e.g., to perform more front-end process development prior to the first irradiations while maintaining the same completion date), one option would be to run two single-cell screening capsules simultaneously in HFIR: one capsule would include TRIZO fuel compacts, and the other would contain UO_2^* compacts (based upon the available data, UO_2^*-ZrC overcoat would probably be used in the fuel compacts with loose UO_2^*-ZrC buffer particles included in piggyback samples). These accelerated irradiations in HFIR could be completed approximately a year sooner than the less accelerated planned tests in ATR.

If the ATR were not available for use on this program for whatever reason, the HFR Petten would be a viable alternative. Its test capabilities, including a proven four-cell capsule design, are well established from the extensive successful testing of German fuel spheres.

6.3.3 Post Irradiation Examination of Advanced Fuels

Once again, this incremental advanced fuel program would follow the lead of the base AGR program. It is anticipated that most, if not all, of the PIE facilities could be shared whether they are located at ORNL, INEEL or ANL-W; the obvious exception is the post-irradiation heating facility as discussed in the next section.

6.3.4 Post-Irradiation Accident Test Facilities

Two new post-irradiation heating facilities will need to be constructed to support the planned heating program on the proposed schedule, and they are included in the cost estimate. As discussed previously in Section 5, the heating facilities needed for this program must permit heating irradiated fuel compacts and loose particles in dry helium to 2200 ^{o}C and heating in helium/air and helium/steams mixtures to at least 1400 ^{o}C. Whether the new AGR design will accommodate those test conditions is uncertain at this writing. If it does, then the facility design can simply be replicated for use on this program with an attendant cost savings; if not, then additional design work will be need to be funded by this program.

At least one of these new heating facilities should be constructed at INEEL or ANL-W for the following reasons. It is assumed that the irradiations will be done in ATR at INEEL. Moreover, some of the irradiated fuel compacts will need to be reactivated prior to heating. Assuming that a King furnace is installed in the TRIGA reactor at ANL-W, it would be convenient to locate at least one of the new heating facilities in the vicinity and avoid the necessity of multiple rapid cross-country shipments of irradiated fuel.

Table 6-1. Post Irradiation Examination and Test Facilities

Fuel Isotopes	Component Handled	Type Facility	Primary Exams And Tests	Technical Services	Support Services
Post Irradiation Examination	All irradiated fuel and graphite components	Hot Cells (E)	Disassembly, Materials examination and microscopy (E)	Analytical chemistry (E) and metallurgical services	Waste management operations (E)
Irradiated Fuel Test Facility	All irradiated fuel and graphite components	Hot Cells (E)	Themo-chemical and thermo-physical 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				

Figure 6-1. Building Complex Housing HFIR, ORNL

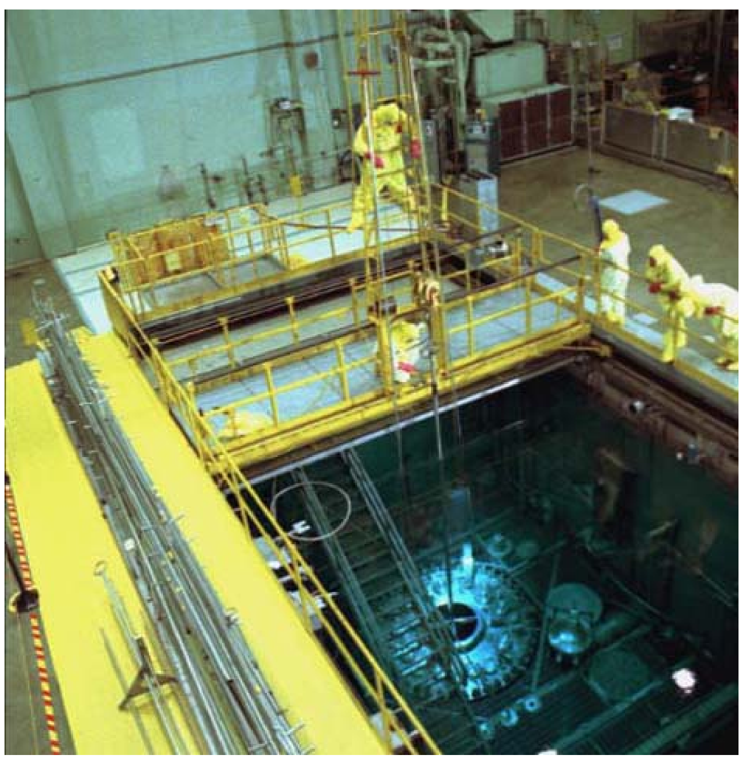


Figure 6-2. HFIR Reactor viewed through pool, ORNL

Figure 6-3. Building, Complex Housing ATR, INEEL

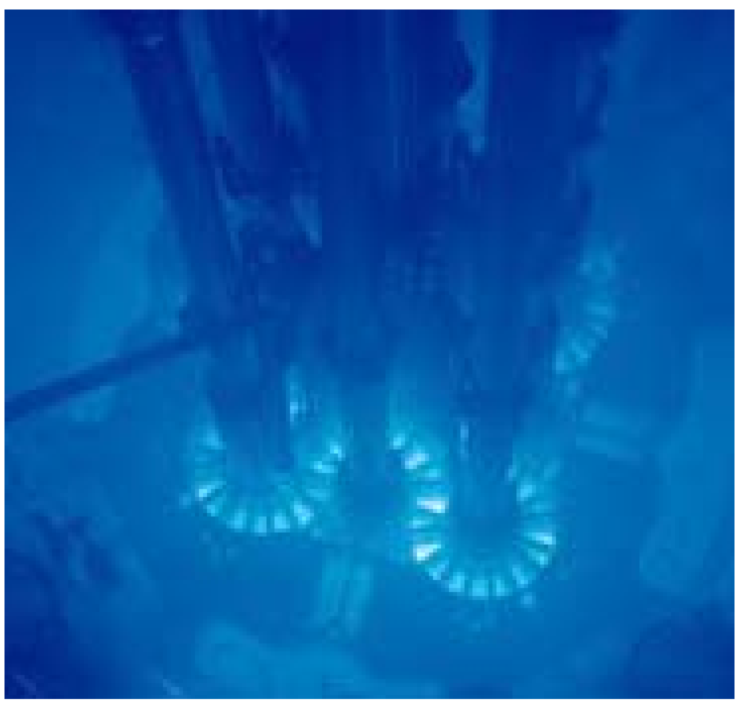


Figure 6-4. ATR Core and Pool, ORNL

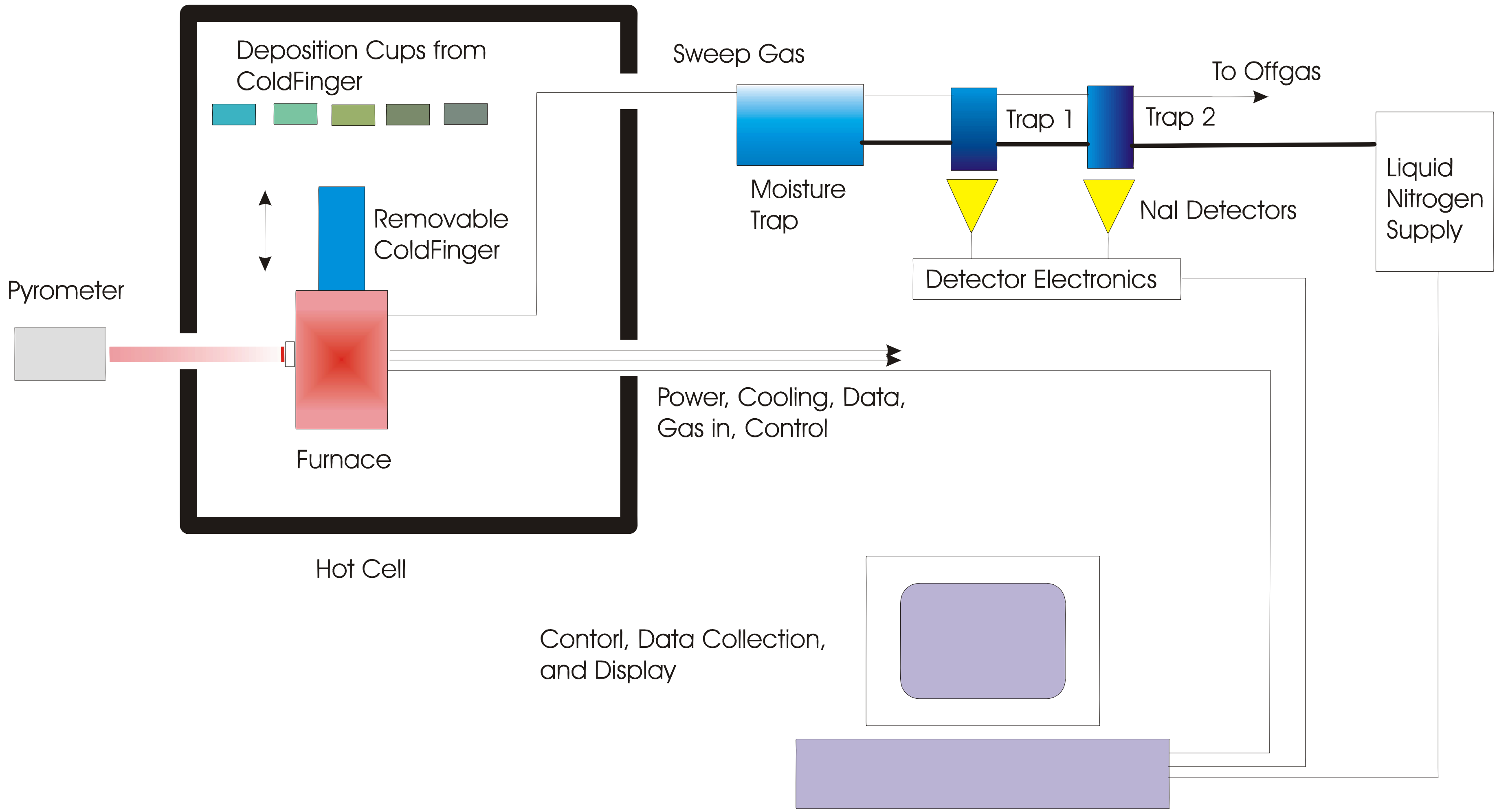


Figure 6-5. Schematic of CCCTF, ORNL

7. Quality Assurance Program

The activities described in this plan 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.

8. Schedule and Cost for the Program

The cost and schedule for the program are arranged in a work breakdown structure (Table 5-1) 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.

An experienced team of coated-particle fuel development experts from ORNL, INEEL, BWXT, and GA developed detailed cost and schedule estimates in 2002-2003 for the AGR fuel program. Since this advanced fuel program consists of similar or identical tasks (with a different fuel) and will utilize similar or identical equipment, the unit durations and costs developed on the AGR program were utilized here, with appropriate adjustments, to develop the detailed cost and schedule estimates. The duration and cost basis for each task in this program are indicated in Table 8-1.

8.1 Detailed Schedule

The summary schedule is shown in Table 1-1, and the detailed schedule is shown in Appendix A. It is consistent with the overall goal of having a qualified advanced particle available at the time of the projected startup of a Demonstration VHTR module in early FY2016. However, it is assumed that at least the first core for the Demonstration Module will use conventional TRISO-coated fuel. In other words, it is assumed that the AGR fuel program will demonstrate that conventional TRISO-coated particles are adequate to meet VHTR performance requirements for operation at least with a 850 ^{o}C core outlet temperature (and, perhaps, to 1000 ^{o}C with core design changes). The durations of key tasks (e.g., capsule irradiation, post-irradiation examination, post-irradiation heating, etc.) were chosen to be consistent with the detailed estimates developed on the AGR program. The planned program continues into FY2016 to complete post-irradiation work on a planned screening capsule with more exotic coatings.

8.2 Detailed Cost Estimate

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 1-2, and details are given in Appendix B.

As summarized in Table 1-2, the total cost of the planned program is about $77 million. As with the task durations, the unit costs for key tasks (e.g., capsule irradiation, etc.) were chosen to be consistent with the detailed cost estimates developed on the AGR program. The cost estimates beyond FY2007 are highly speculative for the following reasons. With the current schedules, a number of key events are scheduled for completion by the end of FY2007. First, the preliminary design phase for the Demonstration Module will have been completed; consequently, the fuel performance requirements and service conditions will be much better established than at this writing. Secondly, the irradiation of the AGR-1 capsule with TRISO-coated UCO fuel will have been completed, giving a better indication of the performance potential of that fuel. Finally, the first two screening capsules planned under this program – VHTR-1 with TRISO-coated UO_2* and VHTR–2 with TRIZO-coated UCO.- will also have completed irradiation. At this point, it is anticipated that both the AGR fuel plan and this plan would be revisited and extensively revised (or, perhaps, even merged).

The cost is dominated by the post-irradiation heating tasks, which account for 31% of the program cost. The costs associated with the irradiation test programs and the post-irradiation work are reasonably well known because of the wealth of experience, although there are significant extrapolations of past experience to the present time frame, involving inflation and an increased oversight for safety.

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 current knowledge, there is good reason to believe that the total cost of this program is approximated closely by this estimate.

Table 8-1. Basis for Duration and Cost Estimate

WBS	Task Title	Duration (Days)	Cost Estimate	Cost Basis	Comments
1.	Fuel Design	4383	$2,845,191		Collector
1.1	Design Data Needs	1552	$224,231		Collector
1.1.1	Issue 0	91	$74,744	3 man-mo at $25K/man-mo	EJ[19]
1.1.2	Issue 1 (Preliminary Design)	91	$74,744	3 man-mo at $25K/man-mo	EJ
1.1.3	Issue 2 (Final Design)	91	$74,744	3 man-mo at $25K/man-mo	EJ
1.2	Fuel Development Plan	1644	$224,231		Collector
1.2.1	Issue 0	91	$74,744	3 man-mo at $25K/man-mo	EJ
1.2.2	Issue 1 (Preliminary Design)	91	$74,744	3 man-mo at $25K/man-mo	EJ
1.2.3	Issue 2 (Final Design)	91	$74,744	3 man-mo at $25K/man-mo	EJ
1.3	Fuel Specifications	1552	$224,231		Collector
1.3.1	Issue 0	91	$74,744	3 man-mo at $25K/man-mo	EJ
1.3.2	Issue 1 (Preliminary Design)	91	$74,744	3 man-mo at $25K/man-mo	EJ
1.3.3	Issue 2 (Final Design)	91	$74,744	3 man-mo at $25K/man-mo	EJ
1.4	Model Development	819	$298,975		Collector
1.4.1	Particle Performance	819	$149,488		Collector
1.4.1.1	Issue 1 (Preliminary Design)	91	$74,744	3 man-mo at $25K/man-mo	EJ
1.4.1.2	Issue 2 (Final Design)	91	$74,744	3 man-mo at $25K/man-mo	EJ
1.4.2	Radionuclide Transport	819	$149,488		Collector

[19] "EJ" denotes engineering judgment based upon extensive past experience in planning and conducting coated-particle fuel development programs.

WBS	Task Title	Duration (Days)	Cost Estimate	Cost Basis	Comments
1.4.2.1	Issue 1 (Preliminary Design)	91	$74,744	3 man-mo at $25K/man-mo	EJ
1.4.2.2	Issue 2 (Final Design)	91	$74,744	3 man-mo at $25K/man-mo	EJ
1.5	Design Methods Validation	3560	$1,873,522		Collector
1.5.1	Methods V & V Plan	91	$74,744	3 man-mo at $25K/man-mo	EJ
1.5.2	Methods Validation Report	1095	$1,798,778	2 eq hd for 3 yr	EJ
2.	Fuel Development	4746	$74,619,172		Collector
2.1	Fuel Process Development	3012	$12,200,071		Collector
2.1.1	Kernel Process Development	2556	$3,297,760		Collector
2.1.1.1	UCO Kernel Optimization	730	$599,593	1 eq hd/yr for 2 yr @ $300K/yr	EJ
2.1.1.2	UO_2^* Kernel Development	730	$1,199,186	2 eq hd/yr for 2 yr @ $300K/yr	EJ
2.1.1.3	Advanced Kernel Process Dev	1825	$1,498,982	1 eq hd/yr for 5 yr @ $300K/yr	EJ
2.1.2	Coating Process Development	2920	$4,796,742		Collector
2.1.2.1	TRISO Coating Process Opt.	730	$599,593	1 eq hd/yr for 2 yr @ $300K/yr	EJ
2.1.2.2	ZrC Coating Process Dev	1095	$2,698,168	3 eq hd/yr for 3 yr @ $300K/yr	EJ
2.1.2.3	Nonconventional Coatings	1825	$1,498,982	1 eq hd/yr for 5 yr @ $300K/yr	EJ
2.1.3	Compact Process Development	730	$599,593	2 eq hd/yr for 2 yr @ $300K/yr	EJ
2.1.4	QC Development	730	$1,199,186	2 eq hd/yr for 2 yr @ $300K/yr	EJ, ZrC
2.1.5	Test Fuel Fabrication	2830	$2,006,993		Collector
2.1.5.1	Screening Tests	2830	$1,048,959		Collector
2.1.5.1.1	VHTR-1 Capsule (UO_2^*)	90	$349,653	Cost for AGR-2 fuel	
2.1.5.1.2	VHTR-2 Capsule (TRIZO)	90	$349,653	Cost for AGR-2 fuel	UCO from AGR?

WBS	Task Title	Duration (Days)	Cost Estimate	Cost Basis	Comments
2.1.5.1.3	VHTR-6 Capsule (Adv. Particles)	90	$349,653	Cost for AGR-2 fuel	Low?
2.1.5.2	Qualification Tests	911	$479,017		Collector
2.1.5.2.1	VHTR-3 Capsule (Ref. Fuel)	180	$159,672	Cost for AGR qualification capsules	
2.1.5.2.2	VHTR-4 Capsule (Ref. Fuel)	180	$159,672	Cost for AGR qualification capsules	
2.1.5.2.3	VHTR-5 Capsule (Ref. Fuel)	180	$159,672	Cost for AGR qualification capsules	
2.1.5.3	Validation Tests	180	$159,672		Collector
2.1.5.3.1	VHTR-7 Capsule (Ref. Fuel)	180	$159,672	Cost for AGR qualification capsules	
2.1.5.4	FP Transport Tests	2646	$319,345		Collector
2.1.5.4.1	VHTR-8 Capsule (UO_2^*/TRIZO)	90	$159,672	Cost for AGR qualification capsules	VHTR-1/2 kernels
2.1.5.4.2	VHTR-9 Capsule (Ref. Fuel)	180	$159,672	Cost for AGR qualification capsules	VHTR-7 kernels
2.1.6	Product Recovery Development	730	$299,796	0.5 eq hd/yr for 2 yr @ $300K/yr	
2.2	Fuel Materials Development	4746	$47,425,524		Collector
2.2.1	Out-of-Pile Characterization	1827	$299,961		Collector
2.2.1.1	Thermochemical Analysis	730	$119,919	0.2 eq hd for 2 yr @ $300K/yr	EJ
2.2.1.2	Material Property Measurements	1096	$180,042	0.2 eq hd for 3 yr @ $300K/yr	EJ
2.2.2	Irradiation Testing	3652	$14,710,714		Collector
2.2.2.1	Screening Tests	3652	$6,817,370		Collector
2.2.2.1.1	VHTR-1 Capsule (UO_2^*)	730	$2,272,457	Cost for AGR-2 (performance test)	
2.2.2.1.2	VHTR-2 Capsule (TRIZO)	730	$2,272,457	Cost for AGR-2 (performance test)	
2.2.2.1.3	VHTR-6 Capsule (Adv. Particles)	730	$2,272,457	Cost for AGR-2 (performance test)	
2.2.2.2	Qualification Tests	1551	$5,917,981		Collector

WBS	Task Title	Duration (Days)	Cost Estimate	Cost Basis	Comments
2.2.2.2.1	VHTR-3 Capsule (Ref. Fuel)	730	$1,972,660	Cost for AGR-5 (qualification test)	
2.2.2.2.2	VHTR-4 Capsule (Ref. Fuel)	730	$1,972,660	Cost for AGR-5 (qualification test)	
2.2.2.2.3	VHTR-5 Capsule (Ref. Fuel)	730	$1,972,660	Cost for AGR-5 (qualification test)	
2.2.2.3	Validation Tests	731	$1,975,363		Collector
2.2.2.3.1	VHTR-7 Capsule (Ref. Fuel)	731	$1,975,363	Cost for AGR-7 (validation test)	
2.2.3	Post-irradiation Examination	3287	$8,199,432		Collector
2.2.3.1	VHTR-1 Capsule (UO_2^*)	365	$1,163,210	Net AGR-2 (performance) PIE cost	
2.2.3.2	VHTR-2 Capsule (TRIZO)	365	$1,163,210	Net AGR-2 (performance) PIE cost	
2.2.3.3	VHTR-3 Capsule (Ref. Fuel)	365	$1,163,210	Net AGR-2 (performance) PIE cost	
2.2.3.4	VHTR-4 Capsule (Ref. Fuel)	365	$1,163,210	Net AGR-2 (performance) PIE cost	
2.2.3.5	VHTR-5 Capsule (Ref. Fuel)	365	$1,163,210	Net AGR-2 (performance) PIE cost	
2.2.3.6	VHTR-6 Capsule (Adv. Particles)	365	$1,163,210	Net AGR-2 (performance) PIE cost	
2.2.3.7	VHTR-7 Capsule (Ref. Fuel)	365	$1,220,171	Net AGR-7 (validation) PIE cost	
2.2.4	Accident Simulation Tests	3607	$24,215,418		Collector
2.2.4.1	Facility Construction	2191	$7,998,568		Collector
2.2.4.1.1	PIH Furnace #1	365	$3,999,284	75% of new CCCTF cost	Less design work
2.2.4.1.2	PIH Furnace #2	365	$3,999,284	75% of new CCCTF cost	Less design work
2.2.4.2	VHTR-1 Capsule (UO_2^*)	320	$2,157,877		Collector
2.2.4.2.1	PIH-1	80	$539,469	AGR-2 (performance) PIH unit cost	
2.2.4.2.2	PIH-2	80	$539,469	AGR-2 (performance) PIH unit cost	
2.2.4.2.3	PIH-3	80	$539,469	AGR-2 (performance) PIH unit cost	

WBS	Task Title	Duration (Days)	Cost Estimate	Cost Basis	Comments
2.2.4.2.4	PIH-4	80	$539,469	AGR-2 (performance) PIH unit cost	
2.2.4.3	VHTR-2 Capsule (TRIZO)	160	$1,281,240		Collector
2.2.4.3.1	PIH-5	80	$539,469	AGR-2 (performance) PIH unit cost	
2.2.4.3.2	PIH-6	80	$539,469	AGR-2 (performance) PIH unit cost	
2.2.4.4	VHTR-3 Capsule (Ref. Fuel)	480	$3,236,816		Collector
2.2.4.4.1	PIH-7	80	$539,469	AGR-2 (performance) PIH unit cost	
2.2.4.4.2	PIH-8	80	$539,469	AGR-2 (performance) PIH unit cost	
2.2.4.4.3	PIH-9	80	$539,469	AGR-2 (performance) PIH unit cost	
2.2.4.4.4	PIH-10	80	$539,469	AGR-2 (performance) PIH unit cost	
2.2.4.4.5	PIH-11	80	$539,469	AGR-2 (performance) PIH unit cost	
2.2.4.4.6	PIH-12	80	$539,469	AGR-2 (performance) PIH unit cost	
2.2.4.5	VHTR-4 Capsule (Ref. Fuel)	160	$1,078,939		Collector
2.2.4.5.1	PIH-13	80	$539,469	AGR-2 (performance) PIH unit cost	
2.2.4.5.2	PIH-14	80	$539,469	AGR-2 (performance) PIH unit cost	
2.2.4.6	VHTR-5 Capsule (Ref. Fuel)	480	$3,236,816		Collector
2.2.4.6.1	PIH-15	80	$539,469	AGR-2 (performance) PIH unit cost	
2.2.4.6.2	PIH-16	80	$539,469	AGR-2 (performance) PIH unit cost	
2.2.4.6.3	PIH-17	80	$539,469	AGR-2 (performance) PIH unit cost	
2.2.4.6.4	PIH-18	80	$539,469	AGR-2 (performance) PIH unit cost	
2.2.4.6.5	PIH-19	80	$539,469	AGR-2 (performance) PIH unit cost	
2.2.4.6.6	PIH-20	80	$539,469	AGR-2 (performance) PIH unit cost	

WBS	Task Title	Duration (Days)	Cost Estimate	Cost Basis	Comments
2.2.4.7	VHTR-6 Capsule (Adv. Particles)	320	$2,157,877		Collector
2.2.4.7.1	PIH-21	80	$539,469	AGR-2 (performance) PIH unit cost	
2.2.4.7.2	PIH-22	80	$539,469	AGR-2 (performance) PIH unit cost	
2.2.4.7.3	PIH-23	80	$539,469	AGR-2 (performance) PIH unit cost	
2.2.4.7.4	PIH-24	80	$539,469	AGR-2 (performance) PIH unit cost	
2.2.4.8	VHTR-7 Capsule (Ref. Fuel)	320	$3,067,287		Collector
2.2.4.8.1	PIH-25	80	$766,822	AGR-7 (validation) PIH unit cost	
2.2.4.8.2	PIH-26	80	$766,822	AGR-7 (validation) PIH unit cost	
2.2.4.8.3	PIH-27	80	$766,822	AGR-7 (validation) PIH unit cost	
2.2.4.8.4	PIH-28	80	$766,822	AGR-7 (validation) PIH unit cost	
2.3	Radionuclide Transport	4380	$14,993,578		Collector
2.3.1	Transport in Reactor Core	3377	$12,444,077		Collector
2.3.1.1	Normal Operation	3377	$8,501,220		Collector
2.3.1.1.1	VHTR-8 (UO_2^*/TRIZO) Irradiation	730	$2,272,457	Cost for AGR-3 (RN release test)	
2.3.1.1.2	VHTR-8 PIE	365	$1,331,096	AGR-3 (RN release test) PIE cost	
2.3.1.1.3	FGR from UO_2^* Kernels	1095	$404,015	AGR task 3.5.1.1 (no HFR B1)	2x fab costs
2.3.1.1.4	FM Diffusivities in UO_2^*	365	$248,831	AGR task 3.5.1.2 (no HFR B1)	
2.3.1.1.5	FM Diffusivities in ZrC	365	$224,847	AGR task 3.5.3.1 (no PIE costs)	
2.3.1.1.6	Diffusivities in Refractory Coatings	365	$224,847	Same as task 2.3.1.1.5 for ZrC	

WBS	Task Title	Duration (Days)	Cost Estimate	Cost Basis	Comments
2.3.1.1.7	VHTR-9 (Ref. Fuel) Irradiation	731	$1,975,363	Cost for AGR-8 (RN validation test)	
2.3.1.1.7	VHTR-9 PIE	365	$1,519,968	AGR-8 (RN validation test) PIE cost	
2.3.1.1.8	H-3 Transport in Core Materials	730	$299,796	0.5 eq hd/yr for 2 yr @ $300K/yr	EJ
2.3.1.2	Accident Conditions	2557	$3,942,857		Collector
2.3.1.2.1	VHTR-8 (UO_2^*/TRIZO) PIH	320	$1,855,617		Collector
2.3.1.2.1.1	PIH-29	80	$463,904	AGR-3 (FP Release) PIH cost	
2.3.1.2.1.2	PIH-30	80	$463,904	AGR-3 (FP Release) PIH cost	
2.3.1.2.1.3	PIH-31	80	$463,904	AGR-3 (FP Release) PIH cost	
2.3.1.2.1.4	PIH-32	80	$463,904	AGR-3 (FP Release) PIH cost	
2.3.1.2.2	FGR from UO_2^* Kernels	365	$248,831	AGR task 3.5.1.2	
2.3.1.2.3	FM Diffusivities in UO_2^*	365	$248,831	AGR task 3.5.2.1	
2.3.1.2.4	FM Diffusivities in ZrC	365	$98,933	AGR task 3.5.2.2	
2.3.1.2.5	Diffusivities in Refractory Coatings	365	$98,933	AGR task 3.5.2.2	
2.3.1.2.6	VHTR-9 (Ref. Fuel) PIH	240	$1,391,712		Collector
2.3.1.2.6.1	PIH-33	80	$463,904	AGR-8 (FP Validation) PIH cost	
2.3.1.2.6.2	PIH-34	80	$463,904	AGR-8 (FP Validation) PIH cost	
2.3.1.2.6.3	PIH-35	80	$463,904	AGR-8 (FP Validation) PIH cost	
2.3.2	Transport in Primary Circuit	1460	$2,549,501		Collector
2.3.2.1	Normal Operation	1096	$1,949,909		Collector
2.3.2.1.1	RN Sorption on VHTR Alloys	1096	$1,350,316	1.5 eq hd/yr for 3 yr @ $300K/yr	EJ

WBS	Task Title	Duration (Days)	Cost Estimate	Cost Basis	Comments
2.3.2.1.2	H-3 Permeation of HX Tubes	730	$599,593	1 eq hd/yr for 2 yr @ $300K/yr	EJ
2.3.2.2	Accident Conditions	730	$599,593		Collector
2.3.2.2.1	Reentrainment from VHTR Alloys	730	$599,593	1 eq hd/yr for 2 yr @ $300K/yr	EJ

9. 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, and sufficient archival samples are selected and preserved.

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.

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

Table 9-1 Deliverables Identified

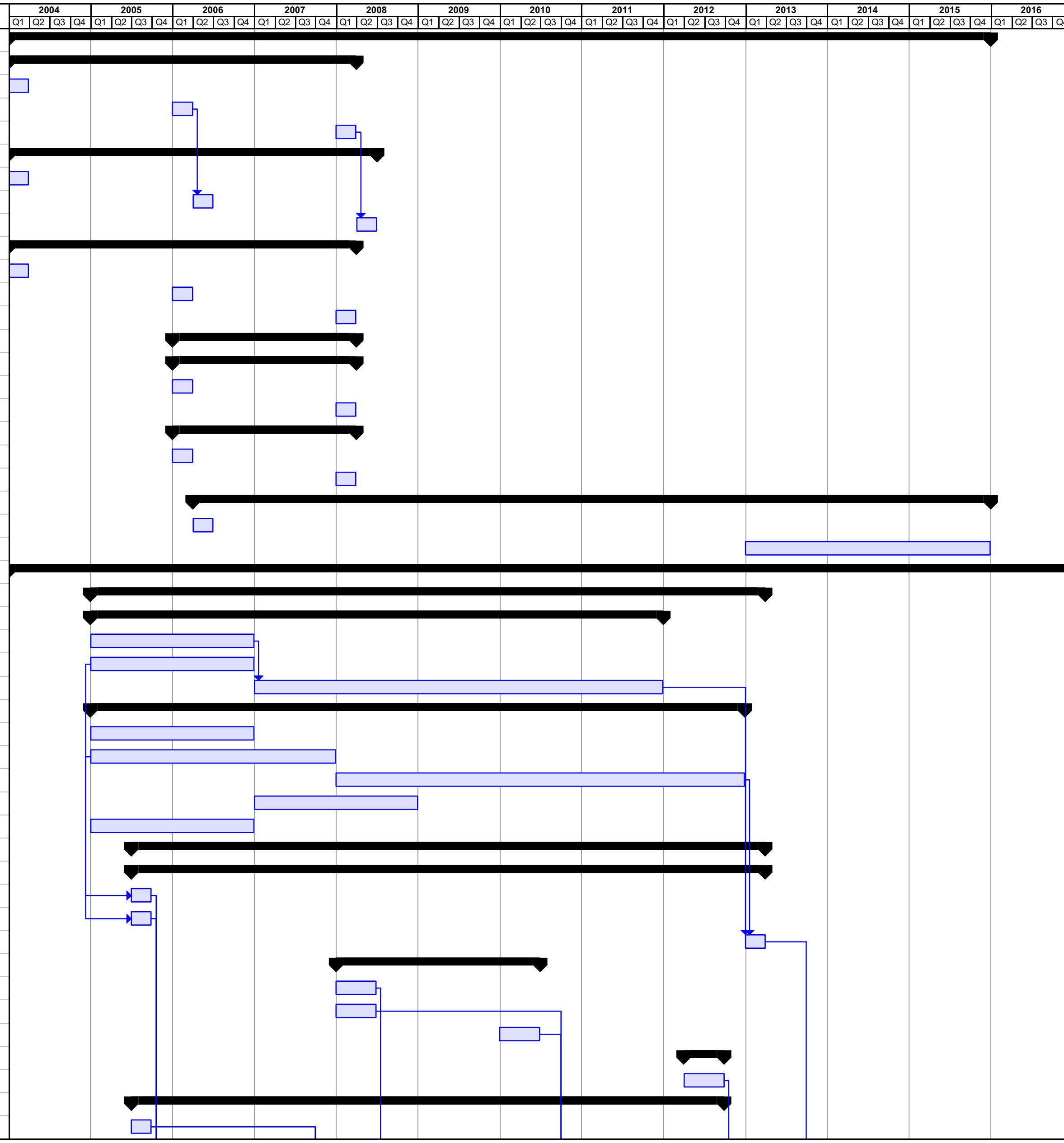
WBS	Task Name	Deliverable	Date
1.	Fuel Design	a. Fuel Development Plan (multiple issues) b. Fuel Specifications (multiple issues) c. Methods V&V Plan d. Methods Validation Report e. Letter reports on program status f. Quarterly progress reports g. Five-year status reports h. Final report i. QA Plan for the program j. Cost and schedule updates (project management)	At the completion of relevant progress, every quarter, every five years, and at project end. The QA plan must be completed before the experimental work begins.
2.	Fuel Development	a. Letter reports on fuel development b. Quarterly progress reports c. Final report on fuel development	At the completion of relevant progress, every quarter, at task completion.
2.1	Fuel Process Development	a. Letter report on UO_2^* particles b. Letter report on TRIZO particles c. UO_2^* particles (test articles) d. TRIZO particles (test articles) e. Letter report on compacting work f. UO_2^* fuel compacts (test articles) g. TRIZO fuel compacts (test articles) h. DTF, laser drilled, and MB particles i. Compacts with DTF, laser drilled & MB particles j. Report on QC methods for ZrC-coated particles k. Process and equipment specifications for reference fuel	At the completion of relevant progress, at completion of task (See schedule, App. A)
2.2	Fuel Materials Development	a. Thermochemical Analysis and Estimation of Performance Report b. Irradiation test specifications (seven capsules) c. PIE specifications (seven capsules) d. Post-irradiation heating specifications (seven capsules)	At the completion of relevant progress, at completion of Task (See schedule, App. A)

WBS	Task Name	Deliverable	Date
		e. Test plans/procedures (seven capsules) f. Multi-cell Irradiation Capsule Design Report (if designed by this program) g. Capsule design reports (nuclear/thermal, seven capsules) h. As-fabricated capsule reports (seven capsules) i. Capsule irradiation reports (seven capsules) j. Capsule PIE reports (seven capsules) k. Post-irradiation heating reports (seven capsules) l. Reference Fuel Selection Report	
2.3	Radionuclide Transport	a. Fission Product Chemical Forms and Implications Report b. Irradiation test specifications (two capsules) c. PIE specifications (two capsules) d. Post-irradiation heating specifications (two capsules) e. Test plans/procedures (two capsules f. Capsule Design Report (if designed by this program) g. Capsule design reports (nuclear/thermal, two capsules) h. As-fabricated capsule reports (two capsules) i. Capsule irradiation reports (two capsules) j. Capsule PIE reports (two capsules) k. Post-irradiation heating reports (two capsules)	At the completion of relevant progress, at completion of Task (See schedule, App. A)

Appendix A: Detailed Development Schedule

Table A-1. Advanced Fuel Development Schedule for VHTR

WBS	Task Name	Total Cost
1	**Fuel Design**	**$2,845,191**
1.1	**Design Data Needs**	**$224,231**
1.1.1	Issue 0	$74,744
1.1.2	Issue 1 (Preliminary Design)	$74,744
1.1.3	Issue 2 (Final Design)	$74,744
1.2	**Fuel Development Plan**	**$224,231**
1.2.1	Issue 0	$74,744
1.2.2	Issue 1 (Preliminary Design)	$74,744
1.2.3	Issue 2 (Final Design)	$74,744
1.3	**Fuel Specifications**	**$224,231**
1.3.1	Issue 0	$74,744
1.3.2	Issue 1 (Preliminary Design)	$74,744
1.3.3	Issue 2 (Final Design)	$74,744
1.4	**Model Development**	**$298,975**
1.4.1	**Particle Performance**	**$149,488**
1.4.1.1	Issue 1 (Preliminary Design)	$74,744
1.4.1.2	Issue 2 (Final Design)	$74,744
1.4.2	**Radionuclide Transport**	**$149,488**
1.4.2.1	Issue 1 (Preliminary Design)	$74,744
1.4.2.2	Issue 2 (Final Design)	$74,744
1.5	**Design Methods Validation**	**$1,873,522**
1.5.1	Methods Verification & Validation Plan	$74,744
1.5.2	Methods Validation Report	$1,798,778
2	**Fuel Development**	**$74,618,762**
2.1	**Fuel Process Development**	**$12,200,892**
2.1.1	**Kernel Process Development**	**$3,298,582**
2.1.1.1	UCO Kernel Optimization	$599,593
2.1.1.2	UO2* Kernel Development	$1,199,186
2.1.1.3	Advanced Kernel Process Dev	$1,499,803
2.1.2	**Coating Process Development**	**$4,796,742**
2.1.2.1	TRISO Coating Process Optimization	$599,593
2.1.2.2	ZrC Coating Process Dev	$2,698,168
2.1.2.3	Nonconventional Coatings	$1,498,982
2.1.3	**Compact Process Development**	$599,593
2.1.4	**QC Development**	$1,199,186
2.1.5	**Test Fuel Fabrication**	**$2,006,993**
2.1.5.1	Screening Tests	**$1,048,959**
2.1.5.1.1	VHTR-1 Capsule (UO2*)	$349,653
2.1.5.1.2	VHTR-2 Capsule (TRIZO)	$349,653
2.1.5.1.3	VHTR-6 Capsule (Adv. Particles)	$349,653
2.1.5.2	Qualification Tests	**$479,017**
2.1.5.2.1	VHTR-3 Capsule (Ref. Fuel)	$159,672
2.1.5.2.2	VHTR-4 Capsule (Ref. Fuel)	$159,672
2.1.5.2.3	VHTR-5 Capsule (Ref. Fuel)	$159,672
2.1.5.3	Validation Tests	**$159,672**
2.1.5.3.1	VHTR-7 Capsule (Ref. Fuel)	$159,672
2.1.5.4	FP Transport Tests	**$319,345**
2.1.5.4.1	VHTR-8 Capsule (UO2*/TRIZO)	$159,672



Project: Advanced Fuel Dev_Rev2.MPP Date: 08/21/03	Task		Milestone		Rolled Up Task		Rolled Up Progress		External Tasks
	Progress		Summary		Rolled Up Milestone		Split		Project Summary

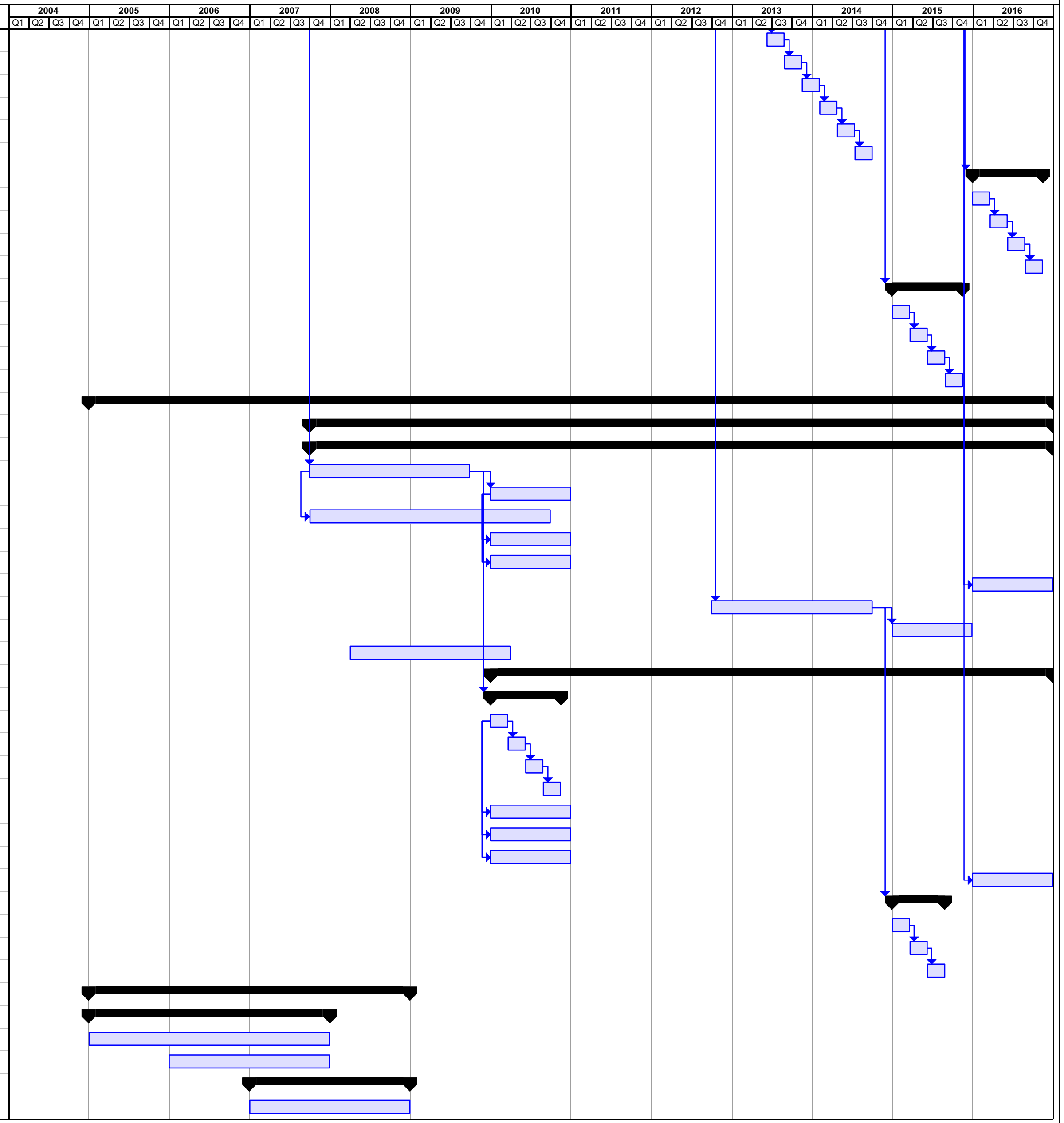
Table A-1. Advanced Fuel Development Schedule for VHTR

WBS	Task Name	Total Cost
2.1.5.4.2	VHTR-9 Capsule (Ref. Fuel)	$159,672
2.1.6	**Product Recovery Development**	$299,796
2.2	**Fuel Materials Development**	**$47,425,524**
2.2.1	**Out-of-Pile Characterization**	**$299,961**
2.2.1.1	Thermochemical Analysis	$119,919
2.2.1.2	Material Property Measurements	$180,042
2.2.2	**Irradiation Testing**	**$14,710,714**
2.2.2.1	**Screening Tests**	**$6,817,370**
2.2.2.1.1	VHTR-1 Capsule (UO2*)	$2,272,457
2.2.2.1.2	VHTR-2 Capsule (TRIZO)	$2,272,457
2.2.2.1.3	VHTR-6 Capsule (Adv. Particles)	$2,272,457
2.2.2.2	**Qualification Tests**	**$5,917,981**
2.2.2.2.1	VHTR-3 Capsule (Ref. Fuel)	$1,972,660
2.2.2.2.2	VHTR-4 Capsule (Ref. Fuel)	$1,972,660
2.2.2.2.3	VHTR-5 Capsule (Ref. Fuel)	$1,972,660
2.2.2.3	**Validation Tests**	**$1,975,363**
2.2.2.3.1	VHTR-7 Capsule (Ref. Fuel)	$1,975,363
2.2.3	**Postirradiation Examination**	**$8,199,432**
2.2.3.1	VHTR-1 Capsule (UO2*)	$1,163,210
2.2.3.2	VHTR-2 Capsule (TRIZO)	$1,163,210
2.2.3.3	VHTR-3 Capsule (Ref. Fuel)	$1,163,210
2.2.3.4	VHTR-4 Capsule (Ref. Fuel)	$1,163,210
2.2.3.5	VHTR-5 Capsule (Ref. Fuel)	$1,163,210
2.2.3.6	VHTR-6 Capsule (Adv. Particles)	$1,163,210
2.2.3.7	VHTR-7 Capsule (Ref. Fuel)	$1,220,171
2.2.4	**Accident Simulation Tests**	**$24,215,418**
2.2.4.1	**Facility Construction**	**$7,998,568**
2.2.4.1.1	PIH Furnace #1	$3,999,284
2.2.4.1.2	PIH Furnace #2	$3,999,284
2.2.4.2	**VHTR-1 Capsule (UO2*)**	**$2,157,877**
2.2.4.2.1	PIH-1	$539,469
2.2.4.2.2	PIH-2	$539,469
2.2.4.2.3	PIH-3	$539,469
2.2.4.2.4	PIH-4	$539,469
2.2.4.3	**VHTR-2 Capsule (TRIZO)**	**$1,281,240**
2.2.4.3.1	PIH-5	$539,469
2.2.4.3.2	PIH-6	$539,469
2.2.4.4	**VHTR-3 Capsule (Ref. Fuel)**	**$3,236,816**
2.2.4.4.1	PIH-7	$539,469
2.2.4.4.2	PIH-8	$539,469
2.2.4.4.3	PIH-9	$539,469
2.2.4.4.4	PIH-10	$539,469
2.2.4.4.5	PIH-11	$539,469
2.2.4.4.6	PIH-12	$539,469
2.2.4.5	**VHTR-4 Capsule (Ref. Fuel)**	**$1,078,939**
2.2.4.5.1	PIH-13	$539,469
2.2.4.5.2	PIH-14	$539,469
2.2.4.6	**VHTR-5 Capsule (Ref. Fuel)**	**$3,236,816**

Project: Advanced Fuel Dev_Rev2.MPP
Date: 08/21/03

Task | Progress | Milestone | Summary | Rolled Up Task | Rolled Up Milestone | Rolled Up Progress | Split | External Tasks | Project Summary

WBS	Task Name	Total Cost
2.2.4.6.1	PIH-15	$539,469
2.2.4.6.2	PIH-16	$539,469
2.2.4.6.3	PIH-17	$539,469
2.2.4.6.4	PIH-18	$539,469
2.2.4.6.5	PIH-19	$539,469
2.2.4.6.6	PIH-20	$539,469
2.2.4.7	**VHTR-6 Capsule (Adv. Particles)**	**$2,157,877**
2.2.4.7.1	PIH-21	$539,469
2.2.4.7.2	PIH-22	$539,469
2.2.4.7.3	PIH-23	$539,469
2.2.4.7.4	PIH-24	$539,469
2.2.4.8	**VHTR-7 Capsule (Ref. Fuel)**	**$3,067,287**
2.2.4.8.1	PIH-25	$766,822
2.2.4.8.2	PIH-26	$766,822
2.2.4.8.3	PIH-27	$766,822
2.2.4.8.4	PIH-28	$766,822
2.3	**Radionuclide Transport**	**$14,992,346**
2.3.1	**Transport in Reactor Core**	**$12,444,077**
2.3.1.1	**Normal Operation**	**$8,501,220**
2.3.1.1.1	VHTR-8 (UO2*/TRIZO) Irradiation	$2,272,457
2.3.1.1.2	VHTR-8 PIE	$1,331,096
2.3.1.1.3	FGR from UO2* Kernels	$404,015
2.3.1.1.4	FM Diffusivities in UO2*	$248,831
2.3.1.1.5	FM Diffusivities in ZrC	$224,847
2.3.1.1.6	FM Diffusivities in Refractory Coatings	$224,847
2.3.1.1.7	VHTR-9 (Ref. Fuel) Irradiation	$1,975,363
2.3.1.1.8	VHTR-9 PIE	$1,519,968
2.3.1.1.9	H-3 Transport in Core Materials	$299,796
2.3.1.2	**Accident Conditions**	**$3,942,857**
2.3.1.2.1	**VHTR-8 (UO2*/TRIZO) PIH**	**$1,855,617**
2.3.1.2.1.1	PIH-29	$463,904
2.3.1.2.1.2	PIH-30	$463,904
2.3.1.2.1.3	PIH-31	$463,904
2.3.1.2.1.4	PIH-32	$463,904
2.3.1.2.2	FGR from UO2* Kernels	$248,831
2.3.1.2.3	FM Diffusivities in UO2*	$248,831
2.3.1.2.4	FM Diffusivities in ZrC	$98,933
2.3.1.2.5	FM Diffusivities in Refractory Coatings	$98,933
2.3.1.2.6	**VHTR-9 (Ref. Fuel) PIH**	**$1,391,712**
2.3.1.2.6.1	PIH-33	$463,904
2.3.1.2.6.2	PIH-34	$463,904
2.3.1.2.6.3	PIH-35	$463,904
2.3.2	**Transport in Primary Circuit**	**$2,548,269**
2.3.2.1	**Normal Operation**	**$1,948,677**
2.3.2.1.1	RN Sorption on VHTR Alloys	$1,349,084
2.3.2.1.2	H-3 Permeation of HX Tubes	$599,593
2.3.2.2	**Accident Conditions**	**$599,593**
2.3.2.2.1	RN Reentrainment from VHTR Alloys	$599,593



Project: Advanced Fuel Dev_Rev2.MPP
Date: 08/21/03

Task | Progress | Milestone | Summary | Rolled Up Task | Rolled Up Milestone | Rolled Up Progress | Split | External Tasks | Project Summary

Appendix B: Detailed Cost Estimate

Table B-1. Cost Estimate for Advanced Fuel Development Program

	2004	2005	2006	2007	2008	2009	2010	2011	2012	2013	2014	2015	2016	Total
Fuel Design														
Design Data Needs														
Issue 0	$74,744													$74,744
Issue 1 (Preliminary Design)			$74,744											$74,744
Issue 2 (Final Design)					$74,744									$74,744
Fuel Development Plan														
Issue 0	$74,744													$74,744
Issue 1 (Preliminary Design)			$74,744											$74,744
Issue 2 (Final Design)					$74,744									$74,744
Fuel Specifications														
Issue 0	$74,744													$74,744
Issue 1 (Preliminary Design)			$74,744											$74,744
Issue 2 (Final Design)					$74,744									$74,744
Model Development														
Particle Performance														
Issue 1 (Preliminary Design)			$74,744											$74,744
Issue 2 (Final Design)					$74,744									$74,744
Radionuclide Transport														
Issue 1 (Preliminary Design)			$74,744											$74,744
Issue 2 (Final Design)					$74,744									$74,744
Design Methods Validation														
Methods Verification & Validation Plan			$74,744											$74,744
Methods Validation Report										$599,593	$599,593	$599,593		$1,798,778
Fuel Development														
Fuel Process Development														
Kernel Process Development														
UCO Kernel Optimization		$299,796	$299,796											$599,593
UO2* Kernel Development		$599,593	$599,593											$1,199,186
Advanced Kernel Process Dev				$299,796	$300,618	$299,796	$299,796	$299,796						$1,499,803
Coating Process Development														
TRISO Coating Process Optimization		$299,796	$299,796											$599,593
ZrC Coating Process Dev		$899,389	$899,389	$899,389										$2,698,168
Nonconventional Coatings					$300,618	$299,796	$299,796	$299,796	$298,975					$1,498,982
Compact Process Development				$299,796	$299,796									$599,593
QC Development		$599,593	$599,593											$1,199,186
Test Fuel Fabrication														
Screening Tests														
VHTR-1 Capsule (UO2*)		$349,653												$349,653
VHTR-2 Capsule (TRIZO)		$349,653												$349,653
VHTR-6 Capsule (Adv. Particles)										$349,653				$349,653
Qualification Tests														
VHTR-3 Capsule (Ref. Fuel)					$159,672									$159,672
VHTR-4 Capsule (Ref. Fuel)					$159,672									$159,672
VHTR-5 Capsule (Ref. Fuel)							$159,672							$159,672
Validation Tests														
VHTR-7 Capsule (Ref. Fuel)									$159,672					$159,672
FP Transport Tests														
VHTR-8 Capsule (UO2*/TRIZO)		$159,672												$159,672
VHTR-9 Capsule (Ref. Fuel)									$159,672					$159,672
Product Recovery Development					$150,309	$149,488								$299,796
Fuel Materials Development														
Out-of-Pile Characterization														
Thermochemical Analysis	$60,124	$59,795												$119,919
Material Property Measurements			$59,959	$59,959	$60,124									$180,042
Irradiation Testing														
Screening Tests														
VHTR-1 Capsule (UO2*)		$286,392	$1,136,228	$849,837										$2,272,457
VHTR-2 Capsule (TRIZO)		$286,392	$1,136,228	$849,837										$2,272,457
VHTR-6 Capsule (Adv. Particles)										$286,392	$1,136,228	$849,837		$2,272,457

Table B-1. Cost Estimate for Advanced Fuel Development Program

	2004	2005	2006	2007	2008	2009	2010	2011	2012	2013	2014	2015	2016	Total
Qualification Tests														
VHTR-3 Capsule (Ref. Fuel)					$494,516	$986,330	$491,814							$1,972,660
VHTR-4 Capsule (Ref. Fuel)							$248,609	$986,330	$737,721					$1,972,660
VHTR-5 Capsule (Ref. Fuel)							$248,609	$986,330	$737,721					$1,972,660
Validation Tests														
VHTR-7 Capsule (Ref. Fuel)									$248,609	$986,330	$740,423			$1,975,363
Postirradiation Examination														
VHTR-1 Capsule (UO2*)					$1,163,210									$1,163,210
VHTR-2 Capsule (TRIZO)					$1,163,210									$1,163,210
VHTR-3 Capsule (Ref. Fuel)							$293,193	$870,017						$1,163,210
VHTR-4 Capsule (Ref. Fuel)										$1,163,210				$1,163,210
VHTR-5 Capsule (Ref. Fuel)										$1,163,210				$1,163,210
VHTR-6 Capsule (Adv. Particles)													$1,163,210	$1,163,210
VHTR-7 Capsule (Ref. Fuel)												$1,220,171		$1,220,171
Accident Simulation Tests														
Facility Construction														
PIH Furnace #1				$3,999,284										$3,999,284
PIH Furnace #2									$3,999,284					$3,999,284
VHTR-1 Capsule (UO2*)														
PIH-1					$539,469									$539,469
PIH-2					$539,469									$539,469
PIH-3					$539,469									$539,469
PIH-4					$539,469									$539,469
VHTR-2 Capsule (TRIZO)						$202,301								$202,301
PIH-5						$539,469								$539,469
PIH-6						$539,469								$539,469
VHTR-3 Capsule (Ref. Fuel)														
PIH-7							$539,469							$539,469
PIH-8							$80,920	$458,549						$539,469
PIH-9								$539,469						$539,469
PIH-10								$539,469						$539,469
PIH-11								$539,469						$539,469
PIH-12								$384,372	$155,097					$539,469
VHTR-4 Capsule (Ref. Fuel)														
PIH-13										$539,469				$539,469
PIH-14										$539,469				$539,469
VHTR-5 Capsule (Ref. Fuel)														
PIH-15										$539,469				$539,469
PIH-16										$539,469				$539,469
PIH-17										$303,452	$236,018			$539,469
PIH-18											$539,469			$539,469
PIH-19											$539,469			$539,469
PIH-20											$539,469			$539,469
VHTR-6 Capsule (Adv. Particles)														
PIH-21													$539,469	$539,469
PIH-22													$539,469	$539,469
PIH-23													$539,469	$539,469
PIH-24													$539,469	$539,469
VHTR-7 Capsule (Ref. Fuel)														
PIH-25												$766,822		$766,822
PIH-26												$766,822		$766,822
PIH-27												$766,822		$766,822
PIH-28												$766,822		$766,822
Radionuclide Transport														
Transport in Reactor Core														
Normal Operation														
VHTR-8 (UO2*/TRIZO) Irradiation				$286,392	$1,139,341	$846,724								$2,272,457
VHTR-8 PIE							$1,331,096							$1,331,096
FGR from UO2* Kernels				$33,628	$135,252	$134,231	$100,904							$404,015

Table B-1. Cost Estimate for Advanced Fuel Development Program

	2004	2005	2006	2007	2008	2009	2010	2011	2012	2013	2014	2015	2016	Total
FM Diffusivities in UO2*							$248,831							$248,831
FM Diffusivities in ZrC							$224,847							$224,847
FM Diffusivities in Refractory Coatings													$224,847	$224,847
VHTR-9 (Ref. Fuel) Irradiation									$248,609	$986,330	$740,423			$1,975,363
VHTR-9 PIE												$1,519,968		$1,519,968
H-3 Transport in Core Materials					$112,526	$149,898	$37,372							$299,796
Accident Conditions														
VHTR-8 (UO2*/TRIZO) PIH														
PIH-29							$463,904							$463,904
PIH-30							$463,904							$463,904
PIH-31							$463,904							$463,904
PIH-32							$463,904							$463,904
FGR from UO2* Kernels							$248,831							$248,831
FM Diffusivities in UO2*							$248,831							$248,831
FM Diffusivities in ZrC							$98,933							$98,933
FM Diffusivities in Refractory Coatings													$98,933	$98,933
VHTR-9 (Ref. Fuel) PIH														
PIH-33												$463,904		$463,904
PIH-34												$463,904		$463,904
PIH-35												$463,904		$463,904
Transport in Primary Circuit														
Normal Operation														
RN Sorption on VHTR Alloys		$449,695	$449,695	$449,695										$1,349,084
H-3 Permeation of HX Tubes			$299,796	$299,796										$599,593
Accident Conditions														
RN Reentrainment from VHTR Alloys				$299,796	$299,796									$599,593
Total	$284,355	$4,639,419	$6,228,537	$8,627,206	$8,470,257	$4,147,503	$7,057,142	$5,903,599	$6,745,362	$7,996,046	$5,071,093	$8,648,568	$3,644,867	$77,463,953