PC-000513
Revision 0

Development Plan for Advanced High Temperature Coated-Particle Fuels

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-000513	0

TITLE: Development Plan for Advanced High Temperature Coated-Particle Fuels

CM APPROVAL/ DATE	REV.	PREPARED BY	APPROVALS RESOURCE/ SUPPORT	APPROVALS PROJECT	REVISION DESCRIPTION/ W.O. NO.
✓ Mike Batobato cn=Mike Batobato, o=Issued, ou=Configuration Management, c=US Signature Valid 2004.01.20 11:04:51 -08'00'	0	D. L. Hanson J. J. Saurwein	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

PAGE ii OF *

AAA 107 (08/2001)

LIST OF EFFECTIVE PAGES

Page Number	**Number of Pages**	**Revision**
i through xii	12	0
1-1 through 1-4	4	0
2-1 through 2-27	27	0
3-1 through 3-27	27	0
4-1 through 4-17	17	0
5-1 through 5-6	6	0
6-1 through 6-36	36	0
7-1 through 7-10	10	0
8-1 through 8-1	1	0
9-1 through 9-10	10	0
10-1 through 10-3	3	0
A-i through A-3	4	0
B-i through B-3	4	0
TOTAL NO. OF PAGES:	161	

Table of Contents

Acronyms and Abbreviations xii

1. **Executive Summary 1-1**

2. **Introduction and Background 2-1**

 2.1 Purpose and Scope 2-1

 2.2 Programmatic Overview 2-1

 2.3 Background 2-5

  2.3.1 Radionuclide Control in HTGRs 2-5

  2.3.2 Performance Limits of Conventional TRISO-Coated Particles 2-8

   2.3.2.1 As-Manufactured Fuel Attributes 2-9

   2.3.2.2 Kernel Chemistry 2-9

   2.3.2.3 Particle Failure Mechanisms 2-9

  2.3.3 Challenges for VHTR Application of Coated-Particle Fuels 2-12

   2.3.3.1 Accelerated Thermochemical Reactions 2-13

   2.3.3.2 Mechanical Effects of Higher Fuel Temperatures 2-13

   2.3.3.3 Increased Fission Product Transport Rates 2-13

 2.4 Development Goals for High Temperature Fuels 2-14

  2.4.1 Improved Coated-Particle Designs 2-14

   2.4.1.1 Optimization of Conventional TRISO-Coated Particle 2-14

   2.4.1.2 New Fuel Kernels 2-15

   2.4.1.3 New Coating Systems 2-15

   2.4.1.4 Getters 2-16

  2.4.2 Improved Fuel Element Designs 2-17

 2.5 Key Assumptions/Development Strategy 2-17

 2.6 Program Coordination and Collaboration 2-18

 2.7 Plan Organization and Content 2-19

 2.8 References for Section 2 2-19

3. **Fuel Development Status 3-1**

 3.1 Fuel Process Development 3-1

  3.1.1 Kernel Fabrication 3-1

  3.1.2 Coated Particle Fabrication 3-2

  3.1.3 Compact Fabrication 3-4

3.1.4 Quality Control Techniques ... 3-5

3.2 Fuel Materials Development ... 3-6

3.2.1 TRISO-Coated Fuels ... 3-6

3.2.2 Advanced Coated-Particle Fuels ... 3-6

3.2.2.1 UO_2* Fuel ... 3-6

3.2.2.2 TRIZO Fuel ... 3-7

3.2.2.3 Other Advanced Fuels ... 3-9

3.2.3 Coated-Particle Performance Modeling ... 3-10

3.2.3.1 Pressure-Vessel Failure ... 3-11

3.2.3.2 Kernel Migration ... 3-12

3.2.3.3 Chemical Attack of the SiC Layer ... 3-12

3.3 Fission Product Transport Technology ... 3-13

3.3.1 Computer Codes ... 3-13

3.3.1.1 Core Performance Codes ... 3-13

3.3.1.2 Support Codes ... 3-14

3.3.2 Material Property Data ... 3-15

3.3.2.1 Radionuclide Release from Fuel Kernels ... 3-15

3.3.2.2 Radionuclide Transport in Coatings ... 3-16

3.3.2.3 Radionuclide Transport in Matrix and Graphite ... 3-19

3.3.3 Validation of Fission Product Transport Methods ... 3-20

3.4 References for Section 3 ... 3-21

4. Provisional Fuel Requirements ... 4-1

4.1 VHTR Fuel Description ... 4-1

4.1.1 Physical Description ... 4-1

4.1.1.1 Fuel Particles ... 4-1

4.1.1.2 Fuel Compacts ... 4-2

4.1.1.3 Fuel Element ... 4-2

4.1.2 Fuel Cycle ... 4-2

4.1.3 VHTR Service Conditions ... 4-2

4.2 VHTR Fuel Requirements ... 4-3

4.3 VHTR Fuel Product Specifications ... 4-4

4.4 References for Section 4 ... 4-6

5. Design Data Needs ... 5-1

5.1 Methodology for Defining Design Data Needs 5-1
5.2 Basis for VHTR Fuel/Fission Product DDNs 5-1
5.3 References for Section 5 5-2
6. Technology Development Programs 6-1
6.1 Fuel Design 6-2
6.1.1 Design Data Needs 6-2
6.1.2 Fuel Development Plan 6-3
6.1.3 Fuel Specifications 6-3
6.1.4 Model Development 6-4
6.1.5 Design Methods Validation 6-4
6.2 Fuel Process Development 6-4
6.2.1 Kernel Fabrication 6-4
6.2.2 Coated Particle Fabrication 6-6
6.2.2.1 UO_2^* Coating Development 6-6
6.2.2.2 ZrC Coating Development 6-6
6.2.2.3 Development of Alternate Fuel Designs 6-8
6.2.3 Compact Fabrication 6-8
6.2.4 Quality Control Techniques 6-9
6.2.5 Test Fuel Fabrication 6-10
6.2.6 Scrap Recovery and Recycle 6-10
6.3 Fuel Materials Development 6-10
6.3.1 Out-of-Pile Characterization 6-11
6.3.2 Irradiation Testing 6-11
6.3.2.1 Screening of Candidate Particle Designs 6-12
6.3.2.2 Qualification Testing of Reference Fuel 6-12
6.3.2.3 Validation Testing of Reference Fuel 6-13
6.3.3 Postirradiation Examination 6-13
6.3.4 Postirradiation Heating Tests 6-16
6.3.4.1 Test Facility Construction 6-16
6.3.4.2 Test Matrix 6-16
6.4 Radionuclide Transport 6-18
6.4.1 Radionuclide Transport in Reactor Core 6-19
6.4.1.1 RN Transport during Normal Operation 6-19

6.4.1.2 RN Transport under Accident Conditions 6-21

6.4.2 RN Transport in Primary Circuit 6-22

6.4.2.1 RN Transport during Normal Operation 6-22

6.4.2.2 RN Transport under Accident Conditions 6-23

6.5 References for Section 6 6-23

7. Facility Requirements for VHTR Fuel Development Plan 7-1

7.1 Test Facility Requirements 7-1

7.2 Description of Test Facilities 7-1

7.2.1 Facilities for the Fabrication of Coated Particle Fuel 7-2

7.2.2 Facilities for Irradiation Testing of Fuel 7-3

7.2.3 Post Irradiation Examination and Test Facilities 7-4

7.2.4 Post-Irradiation Accident Test Facilities 7-5

7.3 Selected Test Facilities for Advanced Fuel Development 7-5

7.3.1 Facilities for the Fabrication of Advanced Particle Fuel 7-5

7.3.2 Facilities for Irradiation Testing of Advanced Fuel 7-5

7.3.3 Post Irradiation Examination of Advanced Fuels 7-6

7.3.4 Post-Irradiation Accident Test Facilities 7-6

8. Quality Assurance Program 8-1

9. Schedule and Cost for the Program 9-1

9.1 Detailed Schedule 9-1

9.2 Detailed Cost Estimate 9-1

10. Deliverables 10-1

Appendix A: Detailed Development Schedule .... A-i

Appendix B: Detailed Cost Estimate .... B-i

List of Tables

Table 1-1. Summary Schedule for Advanced Fuel Development 1-3

Table 1-2. Summary Cost Estimate for Advanced Fuel Development...................................... 1-4

Table 2-1. Properties of Refractory Carbides 2-22

Table 4-1. GT-MHR Reactor Core Parameters 4-7

Table 4-2. GT-MHR Particle Design Parameters 4-8

Table 4-3. GT-MHR Fuel Compact Design Parameters... 4-9

Table 4-4. Provisional Service Conditions for VHTR Fuel.. 4-10

Table 4-5. Coating Integrity Required for VHTR Fuel 4-11

Table 4-6. Provisional Fission Metal Release Limits 4-12

Table 4-7. Fuel Specification for UO_2^* Particles 4-13

Table 4-8. Fuel Specification for TRIZO Particles.. 4-14

Table 5-1. VHTR Fuel/Fission Product Design Data Needs 5-4

Table 6-1. Work Breakdown Structure for Advanced Fuel Development 6-25

Table 6-2. Irradiation Tests to Satisfy VHTR DDNs 6-26

Table 6-3. Planned Irradiation Tests to Develop Advanced VHTR Fuel................................ 6-27

Table 6-4. Capsule Design Conditions 6-29

Table 6-5. Capsule PIE Matrix 6-33

Table 6-6. Postirradiation Compact Heating Test Matrix.. 6-34

Table 7-1. Post Irradiation Examination and Test Facilities 7-7

Table 9-1. Basis for Duration and Cost Estimate 9-3

Table 10-1 Deliverables Identified 10-2

List of Figures

Figure 2-1. VHTR Demonstration Module 2-23
Figure 2-2. VHTR Demonstration Module Master Schedule 2-24
Figure 2-3. TRISO Fuel Particle 2-25
Figure 2-4. Radionuclide Containment System 2-26
Figure 2-5. TRISO Particle Failure Mechanisms 2-27
Figure 4-1. GT-MHR Core Layout 4-15
Figure 4-2. Fuel Element Components 4-16
Figure 4-3. Logic for Deriving Fuel Quality Requirements 4-17
Figure 5-1. Process for Identifying DDNs 5-6
Figure 7-1. Building Complex Housing HFIR, ORNL 7-8
Figure 7-2. HFIR Reactor viewed through pool, ORNL 7-8
Figure 7-3. Building, Complex Housing ATR, INEEL 7-9
Figure 7-4. ATR Core and Pool, ORNL 7-9
Figure 7-5. Schematic of CCCTF, ORNL 7-10

Acronyms and Abbreviations

Acronym	**Definition**
A&AE	activities and associated equipment
AF Plan	[Development Plan for Advanced High Temperature Coated-Particle Fuels]
AFCI	Advanced Fuel Cycle Initiative
AGR	Advanced Gas Reactor
AGR Plan	Advanced Gas Reactor Fuel Development and Qualification Plan
AGR Program	Advanced Gas Reactor Fuel Development and Qualification Program
ANL-W	Argonne National Laboratory – West
ATR	Advanced Test Reactor (INEEL)
BWXT	BWX Technologies [formerly B&W]
CCCTF	Core Conduction Cooldown Test Facility
CVD	chemical vapor deposition
DB-MHR	Deep-Burn Modular Helium Reactor
DDN	Design Data Need
DF	driver fuel
DTF	“designed-to-fail”
DOE	[United States] Department of Energy
EFPD	effective full-power days
EJ	engineering judgment
eq hd	equivalent head
FIMA	fissions per initial metal atom (a measure of burnup in nuclear fuel)
FM	fission [product] metals
FSV	Fort St. Vrain [US HTGR demonstration plant]
GA	General Atomics
Gen-IV	Generation IV
GT-MHR	Gas-Turbine Modular Helium Reactor [prismatic core]
HEU	high-enriched uranium (usually ~93% U-235)
HFIR	High Flux Isotope Reactor
HHT	Hochtemperaturreaktor mit Helium Turbine [German direct-cycle HTR]
HTGR	High Temperature Gas-Cooled Reactor [generic term for reactor type]

Acronym	**Definition**
HPS	helium purification system
HTR	High Temperature Reactor [pebble-bed core]
IHX	intermediate heat exchanger
IMF	inert matrix fuels
IMGA	Irradiated Microsphere Gamma Analysis
INEEL	Idaho National Engineering and Environmental Laboratory
IPyC	inner pyrolytic carbon coating in a TRISO particle
LANL	Los Alamos National Laboratory
LASL	Los Alamos Scientific Laboratory (now LANL)
LEU	low-enriched uranium (<19.9% U-235)
LF	laser failed
LWR	light-water reactor
MB	missing buffer
MHR	Modular Helium Reactor
MHTGR	[steam-cycle] Modular High Temperature Gas-Cooled Reactor
NGNP	Next Generation Nuclear Plant
NNSA	National Nuclear Security Administration [independent agency within DOE]
NPR	New Production Reactor, an application for producing tritium
OPyC	outer pyrolytic carbon coating in a TRISO particle
ORNL	Oak Ridge National Laboratory
PBMR	Pebble Bed Modular Reactor
PGA	particle gas analyzer
PIE	postirradiation examination
PIH	postirradiation heating
P-PyC	"protective" pyrocarbon coating in a TRISO-P particle
PSID	Preliminary Safety Information Document
QC	quality control
SEM	scanning electron microscope
R/B	release rate-to-birth rate ratio [fission gas release measurement]
RIAR	Research Institute of Atomic Reactors
RN	radionuclide
SiC	silicon carbide coating in a TRISO particle

Acronym	**Definition**
TBD	to be determined
TF	transmutation fuel
TRIGA	Test, Research, Isotopes General Atomics [a test reactor]
TRISO	TRi-ISOtropic coated-fuel particle design with three materials in coating system (low-density PyC, high-density PyC, and SiC)
TRU	transuranic
UCO	a mixture of UO_2 and UC_2
UO_2^*-B	UO_2-ZrC buffer
UO_2^*-C	UO_2-ZrC overcoat
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 Development Plan for Advanced High Temperature Coated-Particle Fuels describes a technology development program to provide the technical basis for selecting and qualifying an advanced coated-particle fuel for the VHTR. The plan would 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 (SiC-based) 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. The workscope includes: (1) fuel process development, (2) capsule irradiation tests, (3) postirradiation examinations, and (4) postirradiation heating (accident simulation) tests which would identify, develop and qualify advanced coated-particle fuels capable of meeting anticipated VHTR fuel performance requirements.[1]

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. 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 (e.g., NbC, etc.) would follow as necessary.

Nine irradiation tests, using the multi-cell capsule being designed by the AGR program, and 35 postirradiation heating tests were defined to satisfy these VHTR DDNs. The first two irradiation capsules (VHTR-1 and VHTR-2) and the first series of postirradiation 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[2] and on validating the associated design methods for predicting its performance during normal plant operation and postulated accidents.

The proposed development program is success oriented. If neither of the two leading candidates were to perform sufficiently well in the first two screening capsules, or if the selected reference fuel did not meet anticipated VHTR performance requirements in the first qualification test

---

[1] This umbrella development plan was preceded by a screening plan (PC-000510/0) which focused on the irradiation and accident simulation tests needed to select and qualify an advanced fuel for the VHTR. The scope, schedule and cost elements of this screening plan are included and embellished in this umbrella plan.

[2] In this Plan "reference particle" should be interpreted as shorthand for "reference advanced fuel particle."

(VHTR-3); then significant particle design changes and/or process optimization might be required and the first qualification test repeated. Based upon the available performance data for UO_2^* and for ZrC-coated particles, it is concluded that the greatest programmatic risk is not technical feasibility but rather schedule risk.

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 (SiC-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 °C core outlet temperature (and, perhaps, to 1000 °C with appropriate core design changes). The durations of key tasks (e.g., capsule irradiation, postirradiation examination, postirradiation heating, etc.) were chosen to be consistent with the detailed estimates that were developed on the AGR program. The planned program continues into FY2016 to complete postirradiation 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 $80 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 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).

Table 1-1. Summary Schedule for Advanced Fuel Development

WBS	Task Name	Start	Finish
1	Fuel Design	7/1/04	9/30/15
1.1	Design Data Needs	7/1/04	12/30/07
1.2	Fuel Development Plan	7/1/04	3/31/08
1.3	Fuel Specifications	7/1/04	12/30/07
1.4	Model Development	10/1/05	12/30/07
1.5	Design Methods Validation	1/1/06	9/30/15
2	Fuel Development	7/1/04	9/29/16
2.1	Fuel Process Development	7/1/04	6/27/13
2.1.1	Kernel Process Development	10/1/04	9/30/11
2.1.2	Coating Process Development	10/1/04	9/28/12
2.1.3	Compact Process Development	10/1/06	9/30/08
2.1.4	QC Development	7/1/04	9/30/10
2.1.5	Test Fuel Fabrication	4/1/05	6/27/13
2.1.6	Product Recovery Development	10/1/07	9/29/09
2.2	Fuel Materials Development	7/1/04	9/29/16
2.2.1	Out-of-Pile Characterization	7/1/04	9/30/08
2.2.2	Irradiation Testing	9/28/05	6/30/15
2.2.2.1	Screening Tests	9/28/05	6/30/15
2.2.2.1.1	VHTR-1 Capsule (UO2*)	9/28/05	9/27/07
2.2.2.1.2	VHTR-2 Capsule (TRIZO)	9/28/05	9/27/07
2.2.2.1.3	VHTR-6 Capsule (Adv. Particles)	7/1/13	6/30/15
2.2.2.2	Qualification Tests	6/30/08	6/29/12
2.2.2.2.1	VHTR-3 Capsule (Ref. Fuel)	6/30/08	6/29/10
2.2.2.2.2	VHTR-4 Capsule (Ref. Fuel)	7/1/10	6/29/12
2.2.2.2.3	VHTR-5 Capsule (Ref. Fuel)	7/1/10	6/29/12
2.2.2.3	Validation Tests	9/26/12	9/26/14
2.2.2.3.1	VHTR-7 Capsule (Ref. Fuel)	9/26/12	9/26/14
2.2.3	Postirradiation Examination	12/27/07	9/29/16
2.2.4	Accident Simulation Tests	12/27/06	8/15/16
2.3	Radionuclide Transport	10/1/05	8/25/16
2.3.1	Transport in Reactor Core	10/1/05	8/25/16
2.3.1.1	Normal Operation	10/1/05	9/30/15
2.3.1.1.1	VHTR-8 (UO2*/TRIZO) Irradiatio	10/1/05	9/30/10
2.3.1.1.2	VHTR-9 (Ref. Fuel) Irradiation	6/29/12	9/30/15
2.3.1.2	Accident Conditions	10/1/05	8/25/16
2.3.2	Transport in Primary Circuit	10/1/05	9/30/10
2.3.2.1	Normal Operation	10/1/05	9/30/09
2.3.2.2	Accident Conditions	10/1/08	9/30/10

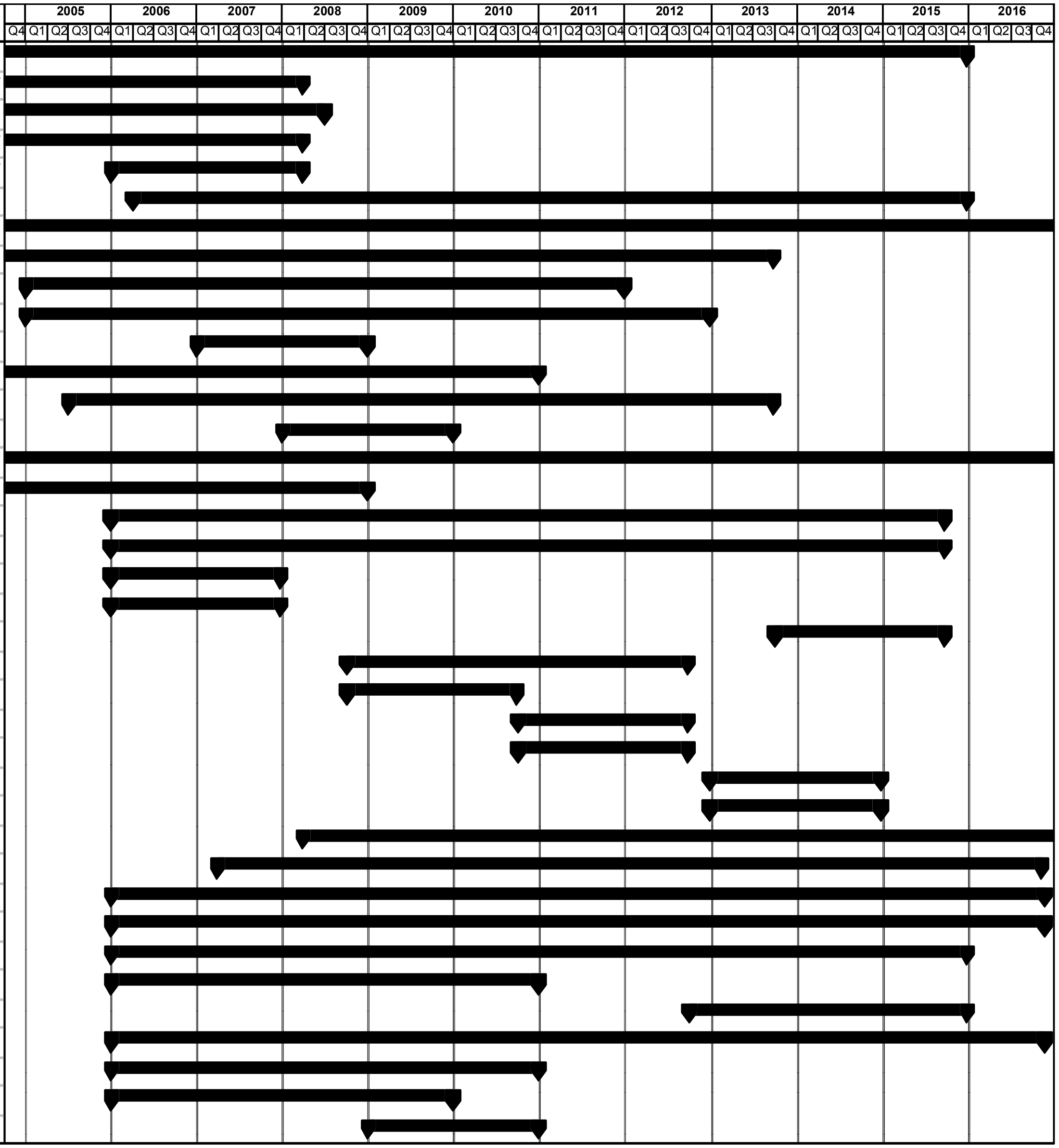


Table 1-2. Summary Cost Estimate for Advanced Fuel Development

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	149	75									374
Model Development			224	149	224									597
Design Methods Validation			75							600	600	600	0	1875
Fuel Development														
Fuel Process Development														
Kernel Development		2164	60	30	30	30	30	30						2374
Coating Development		1900	2039	948	303	300	300	300	774					6864
Compact Development				222	222									444
QC Methods	227	899	673			300	300							2399
Test Fuel Fabrication		899			185	986	554		529	370				3523
Product Recovery					30	30								60
Fuel Materials Development														
Out-of-Pile Characterization	15	60	105	60	60									300
Irradiation Testing														
Screening Tests		19	2272	2254						286	1136	850		6817
Qualification Tests					251		1232	1972	1475					4930
Validation Tests									14	986	976			1976
Postirradiation Examination					1778	548	10	1154		2326		933	1451	8200
Accident Simulation Tests				3046	2835	1558	20	2461	4755	2461	1854	2674	2550	24214
Radionuclide Transport														
Transport in Reactor Core														
Normal Operation					1387	1417	1537	440	249	986	740	1520	225	8501
Accident Conditions							2052	401				1392	99	3944
Transport in Primary Circuit														
Normal Operation			450	450	749	300								1949
Accident Conditions						300	300							600
Total	467	5941	6123	7308	8279	5769	6335	6758	7796	8015	5306	7969	4325	80391

2. Introduction and Background

2.1 Purpose and Scope

This development plan describes the workscope, schedule and cost for a technology program to provide the technical basis for selecting and qualifying an advanced coated-particle fuel capable of meeting anticipated Very High Temperature Reactor (VHTR) fuel performance requirements. The program would 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. This umbrella development plan was preceded by an earlier screening plan (Hanson 2003, PC-000510/0) which focused on the irradiation and accident simulation tests needed to select and qualify an advanced fuel for the VHTR. The scope, schedule and cost elements of this screening plan are included and embellished in this umbrella plan.

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 (SiC-based) 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 (now referred as the Next Generation Nuclear Plant, NGNP) which is projected to begin operation in early FY2016 at the Idaho National Engineering and Environmental Laboratory (INEEL).

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. 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 (e.g., NbC, etc.) would follow as necessary.

2.2 Programmatic Overview

The programmatic context in which this 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 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 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 Gen IV project identified reactor system concepts for producing electricity, which excelled at meeting Gen 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 Project,[3] a project to demonstrate emissions-free nuclear-assisted electricity and hydrogen production by 2015 (e.g., MacDonald 2003a). 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; 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 primary barrier to fission product release from an HTGR core is the fuel particle with its ceramic coatings. Shown schematically in Fig. 2-3, TRISO-coated particles have four coating layers which encapsulate a dense microsphere ("kernel") which contains the fissile and/or fertile materials. The coating layers of a TRISO particle have specialized purposes but, in composite, provide a high integrity pressure vessel which is extremely retentive of fission products. 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 inner pyrocarbon coating (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 IPyC coating is realized during fuel fabrication.

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. The outer pyrocarbon coating (OPyC) layer provides a smooth bonding surface for the production of fuel compacts. The IPyC and the OPyC shrink under irradiation, which produces a compressive stress in the dimensionally stable SiC which compensates for the tensile stress in the SiC induced by the internal gas pressure. These PyC layers also effectively retain fission gases in particles with defective or failed SiC layers.

---

[3] 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.

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 outlet 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 qualifying TRISO-coated (SiC) fuel with a low-enriched uranium (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 ^{o}C (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 as feedstock), 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 in a DB-MHR is accomplished first 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 is a 600 MW(t) direct-cycle GT-MHR with a modified fuel element and a core outlet temperature of 850 °C. 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 replacing SiC) 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 °C and, perhaps, up to 1000 °C (with certain core design changes to limit fuel temperatures). However, as core outlet temperatures are increased to 1000 °C 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 this 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, postirradiation examination (PIE), and postirradiation heating (PIH) tests. As summarized in Section 3, two promising advanced particle designs appear to be more mature than the others (at least based upon information published in the open literature[4]): (1) TRISO-coated UO_2^* (a 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 here 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.

As previously stated, 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 postirradiation heating data before the effects of particle design changes can be reliably determined.

---

[4] There may be classified data on nonconventional coated-particle fuels for various applications, but such data, if they do exist, are not currently available for use on the NGNP program; hence, for planning purposes the possibility of such data is ignored herein.

A relevant example of past experience is the TRISO-P particle (Leikind 1993) which was adopted as the reference particle for the 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 during 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 co-deposited with the porous PyC buffer layer (referred to as UO_2^*-B herein). As discussed in Section 3, 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 Section 3, 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 plan emphasizes obtaining early irradiation and postirradiation 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 Background

2.3.1 Radionuclide Control in HTGRs

A fundamental requirement in the design of any nuclear power plant is the containment and control of the radionuclides produced by the various nuclear reactions; in response, different radionuclide containment systems have been designed and employed for different reactor designs. For modular HTGRs, the approach has been since the early 1980s to design the plant such that the radionuclides would be retained in the core during normal operation and postulated accidents. The key to achieving this safety goal is the reliance upon 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.

This innovative design philosophy - radionuclide containment at the source for all credible plant conditions - has been discussed in numerous publications, but it is perhaps best elaborated in a Preliminary Safety Information Document (PSID) for the 350 MW(t) steam-cycle Modular HTGR (MHTGR) that was submitted to the U.S. Nuclear Regulatory Commission (PSID 1992). This philosophy has been carried forward for all subsequent MHR designs.

The radionuclide containment system for an HTGR, which reflects a defense-in-depth philosophy, is comprised of multiple barriers to limit radionuclide release from the fuel to the environment to insignificant levels during normal operation and a spectrum of postulated accidents. As shown schematically in Fig. 2-4, 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 reactor building/containment structure. The effectiveness of these individual barriers for containing radionuclides depends upon a number of fundamental factors including the chemistry and half-lives of the various radionuclides, the service conditions, and irradiation effects. The effectiveness of the release barriers is also event specific.

The first barrier to fission product release is the fuel kernel itself. Under normal operating conditions, the kernel retains >95% of the radiologically important, short-lived fission gases such as Kr-88 and I-131. However, the effectiveness of a kernel for retaining gases can be reduced at elevated temperatures or if an exposed kernel is hydrolyzed by reaction with trace amounts of water vapor which may be present in the helium coolant (the UO_2 kernel used in PBMR fuel is somewhat less susceptible to hydrolysis effects than is UCO used in the GT-MHR). The retentivity of oxidic fuel kernels for long-lived, volatile fission metals such as Cs, Ag, and Sr is strongly dependent upon the temperature and burnup.

The second - and most important - barrier to fission product release from the core is the ceramic coating system of each fuel particle. In conventional TRISO particles, both the SiC and PyC coatings provide a barrier to the release of fission gases. The SiC coating acts as the primary barrier to the release of metallic fission products because of the low solubilities and small diffusion coefficients of fission metals in SiC; the PyC coatings are partially retentive of Cs at lower temperatures but provide little holdup of Ag and Sr.

With a prismatic core, the fuel-compact matrix and the graphite fuel block collectively are the third release barrier (with a pebble-bed core, the analog is the pebble matrix, including the unfueled outer shell). The fuel-compact matrix is relatively porous and provides little holdup of the fission gases which are released from the fuel particles; however, the matrix is a composite material which has a high content of amorphous carbon, and this constituent is highly sorptive of metallic fission products, especially Sr. While the matrix is highly sorptive of metals, it provides little diffusional resistance to the release of fission metals because of its high interconnected porosity.

The fuel element graphite, which is denser and has a more ordered structure than the fuel-compact matrix, is somewhat less sorptive of the fission metals than the matrix, but it is much more effective as a diffusion barrier than the latter. The effectiveness of the graphite as a release barrier decreases as the temperature increases. Under typical core conditions, the fuel element graphite attenuates the release of Cs from the core by an order of magnitude, and the Sr is essentially completely retained as are the more refractory fission metals and actinides. The extent to which the graphite attenuates Ag release is not nearly as well characterized, and there is some evidence that the retention of Ag by graphite increases as the total system pressure increases (implying gas-phase transport through the interconnected pore structure of the graphite).

Typically, the two dominant sources of fission product release from the core are as-manufactured, heavy metal contamination (i.e., heavy metal outside of the coated particles) and particles whose coatings are defective or fail in service. In addition, the volatile metals (e.g., Cs, Ag, Sr) can, at sufficiently high temperatures for sufficiently long times, diffuse through the SiC coating and be released from intact conventional TRISO particles; however, diffusive release from intact particles is only significant compared to other sources for silver release. Fission products resulting from fissions in heavy-metal contamination outside of the particles are obviously not attenuated by the kernels or coatings, nor are the fission products produced in the kernels of failed particles appreciably attenuated by the failed coatings. In these cases, the fission products must be controlled by limiting the respective sources and by the fuel element graphite in the case of the fission metals and actinides.

The fourth release barrier is the primary coolant pressure boundary. Once the fission products have been released from the core into the coolant, they are transported throughout the primary circuit by the helium coolant. The helium purification system (HPS) efficiently removes both gaseous and metallic fission products from the primary coolant at a rate determined by the gas flow rate through the purification system (the primary purpose of the HPS is to control chemical impurities in the primary coolant). However, for the condensable fission products, the dominant removal mechanism is deposition ("plateout") on the various helium-wetted surfaces in the primary circuit (i.e., the deposition rate far exceeds the purification rate). The plateout rate is determined by the mass transfer rates from the coolant to the fixed surfaces and by the sorptivities of the various materials of construction for the volatile fission products and by their service temperatures. Condensable radionuclides may also be transported throughout the primary circuit sorbed on particulates ("dust") which may be present in the primary coolant; the plateout distribution of these contaminated particulates may be considerably different than the distribution of radionuclides transported as atomic species.

The circulating and plateout activities in the primary coolant circuit are potential sources of environmental release in the event of primary coolant leaks or as a result of the venting of primary coolant in response to overpressuring of the primary circuit (e.g., in response to significant water ingress in a steam-cycle plant). The fraction of the circulating activity lost during such events is essentially the same as the fraction of the primary coolant that is released, although the radionuclide release can be mitigated by pump down through the HPS if the leak rate is sufficiently slow.

A small fraction of the plateout may also be reentrained, or "lifted off," if the rate of depressurization is sufficiently rapid. The amount of fission product liftoff is expected to be strongly influenced by the amount of dust in the primary circuit as well as by the presence of friable surface films on primary circuit components which could possibly spall off during a rapid depressurization.

Other mechanisms which can potentially result in the removal and subsequent environmental release of primary circuit plateout activity are "steam-induced vaporization" and "washoff." In both cases, the vehicle for radionuclide release from the primary circuit is water which has entered the primary circuit. In principle, both water vapor and liquid water could partially remove plateout activity. However, even if a fraction of the plateout activity were removed from the fixed surfaces, there would be environmental release only in the case of venting of the helium/steam mixture from the primary circuit. For all but the largest water ingress events the pressure relief valve does not lift. Moreover, the radiologically important nuclides such as iodine and cesium are expected to remain preferentially in the liquid water which remains inside the primary circuit. The probability of large water ingress with a direct-cycle plant is much lower than for a steam-cycle plant because with the former the secondary water pressures are lower than the primary He pressures, and the heat-exchanger tubes operate at much lower temperatures during normal operation.

The reactor building/containment structure is the fifth barrier to the release of radionuclides to the environment. Its effectiveness as a release barrier is highly event-specific. The vented low pressure containment (VLPC) may be of limited value during rapid depressurization transients; however, it is of major importance during longer term, higher risk, core conduction cooldown transients during which forced cooling is unavailable. Under such conditions, the natural removal mechanisms occurring in the VLPC, including condensation, fallout and plateout, serve to attenuate the release of condensable radionuclides, including radiologically important iodines, by at least an order of magnitude

Although still in preconceptual design, the NGNP radionuclide containment system will undoubtedly include the functional equivalent of each of these release barriers, and credit will likely be taken for each of these barriers when demonstrating compliance with the top-level radionuclide control requirements which are imposed upon the design. However, in this plan, the emphasis is on characterizing the performance attributes of the first two release barriers - kernels and coatings - of candidate advanced particle designs. In large measure, it will be the responsibility of the AGR program to determine the performance characteristics of the other release barriers.

2.3.2 Performance Limits of Conventional TRISO-Coated Particles

During the past four decades of coated-particle fuel development, demonstration and utilization, a number of mechanisms have been identified - and quantified - which can compromise the capability of the coated fuel particles to retain radionuclides (i.e., functional failure of the particle). A considerable number of documents have been prepared on the topic of coated particle failure mechanisms; IAEA-TECDOC-978 provides a good summary of these mechanisms along with an extensive bibliography.

It is noteworthy that the in-service performance of coated particles can be strongly influenced by their as-manufactured attributes and that the choice of kernel composition also influences performance; those aspects are briefly mentioned before the individual failure mechanisms are reviewed.

2.3.2.1 As-Manufactured Fuel Attributes

The in-service coating performance may be profoundly influenced by the as-manufactured coating material properties (e.g., the performance of PyC coatings under irradiation is strongly influenced by the degree of anisotropy). Ideally, it would be desirable to rely exclusively on product specifications to control the as-manufactured attributes of coated-particle fuel (especially if there were multiple independent fuel suppliers). Unfortunately, the mechanistic understanding of particle performance and the available quality control (QC) methods are not yet adequate to guarantee with sufficient confidence the required in-service performance based upon measured product attributes; consequently, it is necessary to supplement the product specifications with process specifications, at least for the foreseeable future. The reader is referred to the specialist literature (e.g., Bullock 1994, Petti 2002) for a more comprehensive treatment of this complicated, and often confusing and contradictory, subject.

2.3.2.2 Kernel Chemistry

The kernel composition can also have a major impact on coating performance. With stoichiometric oxide fuel particles (including UO_2), carbon monoxide is produced from excess oxygen liberated upon fissioning of the heavy metal reacting with the carbonaceous buffer layer because the fission products in the aggregate are thermochemically incapable of binding all of the liberated oxygen. Significant CO formation during irradiation is undesirable for the following reasons: (1) it contributes to, and may dominate at high burnup, the internal gas pressure which must be contained by the coating system; (2) it appears to be a prerequisite for kernel migration in oxidic kernels, and (3) CO may corrode the SiC coatings at the high temperatures characteristic of core heatup accidents.

One effective way to control the CO pressure within uranium fuel particles (and therefore kernel migration) is to provide a carbide phase within the kernel that can be oxidized in preference to elemental carbon (e.g., Homan 1977). Each U-235 fission in UO_2 leads to fission products that, at maximum may combine with only ~1.62 of the two oxygen atoms released, leaving, at a minimum, 0.38 atoms available to oxidize other materials, such as carbon or carbides. Oxycarbide fuel (UC_xO_{2-x}) is designed such that UC_2 is converted to UO_2 from the reaction with O_2 liberated by fissioning of UO_2 (the optimal carbide fraction depends upon the design burnup). The oxygen potential is fixed by the UC_2/UO_2 equilibrium, meaning that rare-earth fission products will form oxides and the fission products zirconium, strontium, europium, and barium will form carbides. Another alternative would be to add a getter, such as zirconium, to the system which is preferentially oxidized instead of the carbon of the buffer layer

2.3.2.3 Particle Failure Mechanisms

The following failure mechanisms have been identified as capable of causing partial or total failure of the TRISO coating system under irradiation and during postulated accidents; they are shown schematically in Fig. 2-5. Phenomenological performance models, typically inspired by first principles and correlated with experimental data, have been developed to model each of these mechanisms. Design methods incorporating these models have been developed to predict fuel performance and fission product release from the reactor core to the primary coolant; these design methods are described in Section 3.

1. Coating damage during fuel manufacture, resulting in heavy metal (HM) contamination on coating surfaces and in the fuel-compact matrix.
2. Pressure vessel failure of standard particles (i.e., particles without manufacturing defects).
3. Pressure vessel failure of particles with defective or missing coatings;
4. Irradiation induced failure of the IPyC coating and potential SiC cracking;
5. Irradiation induced failure of the OPyC coating;
6. Failure of the SiC coating due to kernel migration in the presence of a temperature gradient;
7. Failure of the SiC coating by fission product/SiC interactions;
8. Failure of the SiC coating by thermal decomposition;
9. Failure of the SiC coating due to heavy-metal dispersion in the IPyC coating.

Brief descriptions of these failure mechanisms are provided below. The reader who is not familiar with these particle failure mechanisms is encouraged to consult the pertinent references (e.g., IAEA-TECDOC-978). These TRISO failure mechanisms, or their functional equivalent, are largely generic and may occur in advanced fuels as well, especially those mechanisms related to the pyrocarbon layers, although the rates and functional dependencies may be different. In some cases, certain advanced fuels may be immune to a particular TRISO failure mechanism which can be one of their chief attractions. For example, the SiC coating in TRISO particles can be corroded by certain fission metals, principally Pd, at high temperature; however, ZrC coatings are apparently not corroded by Pd (see Section 3).

The first mechanism listed above – as-manufactured heavy-metal contamination - is not an in-service failure mechanism *per se* but rather an extreme case of as-manufactured coating defects whereby trace amounts of heavy metal are not encapsulated by a single intact coating layer (analogous to "tramp uranium" in LWR fuel). Modern fuel product specifications only allow small fractions of HM contamination ($\sim 10^{-5}$ is typical); nevertheless, it is an important source of fission product release.

As elaborated below, the next four failure mechanisms are structural/mechanical mechanisms, and the latter four are thermochemical mechanisms. Before addressing individual mechanisms, it should be noted that they can occur in isolation or in combination. As an example of the former, a particle whose SiC coating has thermally decomposed at very high temperature does not typically undergo pressure-vessel failure of the PyC coatings; apparently, the PyC becomes sufficiently porous that the internal gas pressure is relieved (an example of "leak-before-break"). In contrast, a particle which experiences irradiation-induced failure of its IPyC layer and/or OPyC layer has a much greater probability of experiencing complete pressure-vessel failure of the remaining coatings (which is what happened with the TRISO-P particle).

The coating system serves as a multi-shell pressure vessel which contains the internal gas pressure imposed by the fission gases and CO. Pressure vessel failure occurs when the stresses in the SiC layer exceeds the strength of the layer. The fraction of particles with a failed SiC coating is calculated using Weibull statistical strength theory (which represents a distribution of SiC strengths within the particle population).

Shrinkage of the pyrocarbon layers during irradiation is a favorable attribute, in terms of the compressive forces applied to the SiC layer, as long as the pyrocarbon layers remain intact. However, pyrocarbon shrinkage produces tensile stresses in the pyrocarbon layers themselves, which can lead to failure of these layers. The strains, stresses and creep generated in the pyrocarbon layers are complex functions of fast neutron fluence, irradiation temperature, and coating material properties. Cracking and differential shrinkage of the PyC layers can impose high local stresses on the SiC layer, depending on the local bond strength between the PyC and SiC layers, which can lead to through wall SiC cracks.

With a properly designed fuel particle manufactured to appropriate product and process specifications, the failure probability of standard, or defect free, TRISO particles (Mechanism 2) is negligibly small compared to other sources of failure. However, if one or more coating layers are missing or defective (Mechanism 3), the pressure-vessel failure probability is much higher. Particles with missing or severely undersized buffer layers are particularly prone to this failure mechanism. Pressure-vessel failure is controlled by proper particle design and by product specifications which limit the allowable fractions of coating defects. Process specifications are also required to assure that the coating layers, especially the PyC layers, have the requisite material properties.

The irradiation performance of PyC coatings is dependent upon the anisotropy because anisotropic PyC coatings shrink excessively under neutron irradiation and may fail as a consequence. For IPyC coatings the irradiation-induced failure mode (Mechanism 4) is typically radial cracks which serve as stress risers for the SiC layer which may result in cracking of the SiC layer or pressure-vessel failure of the entire coating system and may serve as pathways for fission product release and for corrosive agents, including CO in UO_2 particles, to attack the SiC coating.

For OPyC coatings, excessive irradiation-induced shrinkage results in high failure rates (Mechanism 5), thereby eliminating a compressive stress imposed upon the SiC coating which serves to counteract a tensile stress component induced by the internal gas pressure. For OPyC coatings, there is a complicating factor as a result of possible matrix/coating interaction. If the compact matrix, which shrinks considerably under irradiation, is too tightly bonded to the OPyC coating, it will tear the OPyC coating off the SiC coating. Open surface porosity in the OPyC layer encourages an excessively strong bond with the matrix; hence, it must be controlled.

The above mechanical/structural failure mechanisms are typically strongly dependent upon fast fluence and, to a lesser degree, on temperature and burnup. The following thermochemical failure mechanisms are strongly temperature dependent with a lesser dependence on fast fluence and burnup.

Local fuel temperatures and temperature gradients across the fuel compact can be relatively high when the reactor is producing power. Under these conditions, oxide and carbide fuel kernels can migrate up the temperature gradient. This phenomenon is often referred to as the "amoeba effect" and can lead to complete failure of the coating system, especially if the kernel migrates to the point of contacting the SiC coating (Mechanism 6). For carbide kernels, migration is caused by solid-state diffusion of carbon to the cooler side of the kernel. For oxide kernels, migration may be caused by carbon diffusion or gas-phase diffusion of CO or other gaseous carbon compounds. Empirically, it is observed that if CO formation is suppressed, kernel migration is also suppressed.

Noble metals (e.g., Ru, Rh, Pd, and Ag) are produced with relatively high yields during fission of uranium and plutonium fuels. During irradiation, the thermochemical conditions are not conducive for these elements to form stable oxides, and they can readily migrate out of the fuel kernel, regardless of its composition. Reactions of SiC with Pd to form palladium silicides have been observed during PIEs of TRISO fuel (Mechanism 7). Although the quantity of fission Pd is small compared with the mass of the SiC layer, the reaction is highly localized and complete penetration ("worm holes") of the SiC layer can occur if high temperatures are maintained for long periods of time. The cumulative fission yield of long-lived and stable Pd isotopes in Pu-239 is more than an order of magnitude higher than that in U-235; consequently, this failure mechanism may be more of a concern for LEU and Pu fuels than for HEU fuels.

At very high temperatures, SiC will decompose into its constituent elements; the silicon vaporizes, leaving a porous carbon structure (Mechanism 8). The coating system remains ostensibly intact. The PyC coatings do not typically undergo pressure-vessel failure; apparently, the PyC becomes sufficiently porous that the internal gas pressure is relieved. Thermal decomposition is not an important contributor to fuel failure at normal operating temperatures. However, relatively high failure rates can occur if temperatures higher than 1700 to 1800 ^{o}C are maintained for extended periods of time, and thermal decomposition of SiC occurs rapidly at temperatures above 2000 ^{o}C.

Heavy metal dispersion results when a defective or porous IPyC layer allows HCl produced during the SiC coating deposition to react with heavy metal in the fuel kernel to form volatile heavy-metal chlorides which are in turn transported out of the kernel into the buffer and IPyC layers. Particles with heavy metal dispersed in the buffer and IPyC are observed to exhibit enhanced SiC attack by fission products and SiC coating failure (Mechanism 9). With proper product and process specifications for the IPyC layer, heavy-metal dispersion is reduced to insignificant levels.

2.3.3 Challenges for VHTR Application of Coated-Particle Fuels

The VHTR Demonstration Module will be designed for a core outlet temperature of 1000 ^{o}C for efficient production of hydrogen and electricity (MacDonald 2003a). The peak fuel temperature in the commercial GT-MHR with an 850 ^{o}C core outlet temperature is expected to be ~1250 ^{o}C for normal operation and <1600 ^{o}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 ^{o}C core outlet temperature. Core design changes have been identified that should permit increased core outlet temperatures without a proportionate increase in peak fuel temperatures during normal operation (e.g., MacDonald 2003b); however, some increases in the average fuel and graphite temperatures must be expected since the average temperatures largely track the coolant temperatures. For core heatup accidents, a 150 ^{o}C increase in core outlet temperature translates into about a 50 °C increase in peak fuel temperature (e.g., MacDonald 2003b).

These higher fuel temperatures in a VHTR core will place an added burden on the fuel, and these higher temperatures are the primary motivation for the development of advanced coated-particle fuels. The implications of higher temperatures on fuel performance are summarized below.

2.3.3.1 Accelerated Thermochemical Reactions

The principal thermochemical failure mechanisms – kernel migration, fission product/SiC interactions, and thermal decomposition – are exponentially temperature dependent, and the activation energies are typically large (Myers 1987); the uncertainties in the kinetics of the fission product/SiC reactions are large. The temperature dependence of HM dispersion-induced failure of the IPyC has not been well characterized. For previous core designs with outlet temperatures of 700 – 850 °C, the contribution of these thermochemical failure mechanisms to the core–average failure fraction has been negligible. Their importance for a core design with a 1000 °C outlet temperature will be largely determined by how successful the core designers are in implementing design changes to control peak fuel temperatures. Since the allowable core-average failure fraction is anticipated to be $\leq 1 \times 0^{-4}$, significant thermochemical failure can only be tolerated in a very small volume of the core.

2.3.3.2 Mechanical Effects of Higher Fuel Temperatures

The mechanical properties of PyC and SiC coatings are complex functions of fast neutron fluence and temperature; in general, the mechanical properties degrade with increasing temperature in the temperature range of interest for VHTR cores (Ho 1993). Pyrocarbon coatings shrink under irradiation; this shrinkage induces tensile stresses in both the IPyC and OPyC layers since the SiC is dimensionally stable (e.g., Leikind 1993). If the stresses exceed the tensile strength of the PyC layer which decreases with increasing temperature, the coating will fail. The failure mode is typically radial cracking; these radial cracks serve as stress risers at the PyC/SiC interfaces, and the cracks may propagate into to the SiC layer leading to complete coating failure. An important mitigating factor is that PyC layers also undergo irradiation-induced creep, and the creep rates increase significantly with increasing temperature. In the aggregate, PyC failure probabilities may well decrease with increasing temperature; however, the uncertainties in the creep rates are large (Ho 1993).

As stated above, SiC is essentially dimensionally stable under HTGR core conditions; however, its strength is a function of temperature. The available data are conflicting (Ho 1993). Early data indicated that the strength of high-density β-SiC increased monotonically up to at least ~1400 °C; more recent data indicated a slight decrease in strength from 600 to 1000 °C, with a rapid decrease thereafter. More data are needed to resolve this discrepancy.

2.3.3.3 Increased Fission Product Transport Rates

For a given particle failure fraction, the attendant fission product release rates from the core will increase with increasing fuel temperature (e.g., Myers 1987). The release rates of fission gases, including radiologically important I-131, from fuel kernels are exponentially temperature dependent, but the activation energies are modest; the release rates of volatile fission metals, including Cs and Sr isotopes, from oxidic fuel kernels are also exponentially temperature dependent, but the activation energies are large. The SiC coating retains the volatile fission metals except for 250-day Ag-110m which is rapidly released from intact conventional TRISO particles at elevated temperatures (say, >1100 ^{o}C); the activation energy is large. The fuel-compact matrix and fuel-element graphite also become less effective barriers to the release of volatile fission metals (Ag, Cs, Sr) from the core as their service temperatures increase. However, the more refractory fission metals (e.g., Zr, Ce, etc.) and the actinides will still be quantitatively retained by the fuel-element graphite.

2.4 Development Goals for High Temperature Fuels

As previously stated, 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 (SiC-based) 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 INEEL.

Nevertheless, it is appropriate to review and evaluate all the available options for the development of advanced coated particle fuels with higher temperature capabilities than conventional TRISO particles; such a review and evaluation would be especially important if the early screening irradiations of UO_2^* and TRIZO-coated UCO were to prove disappointing. Even if both of these advanced particles were to be demonstrated to offer high-temperature performance advantages compared to conventional TRISO particles, they both have known performance limitations. For example, the SiC coating in UO_2^* will degrade rapidly above ~2000 ^{o}C, and ZrC coatings are less oxidation resistant than SiC coatings. Consequently, there is an incentive at least to evaluate more exotic coated particle designs that might tolerate even higher temperatures, thereby permitting higher core power densities (i.e., higher core thermal powers) and/or core outlet temperatures >1000 ^{o}C.

A spectrum of improved coated-particle designs can be envisioned ranging from the optimization of conventional TRISO particles to the development of new kernels and new coating systems. This development plan includes in its latter phase a screening irradiation capsule containing several nonconventional coated-particle designs; the specific particle designs will be identified later as part of the proposed program.

2.4.1 Improved Coated-Particle Designs

2.4.1.1 Optimization of Conventional TRISO-Coated Particle

TRISO fuel particles have been under development for almost four decades, and dramatic improvements have been made in the as-manufactured fuel quality and in-service performance. The German LEU UO_2 particle, which has performed superbly both under irradiation and in accident simulation tests, represents the current state-of-the-art. The combination of the German TRISO coating system with a dense UCO kernel should offer performance advantages by suppressing CO formation at high burnups. Nevertheless, the opportunity for still further optimization of the conventional TRISO particle is evident. For example, it would be highly desirable to optimize the coating process conditions to produce a more isotropic PyC coating. In addition, the potential for producing a superior SiC coating from a non-halide based feedstock is intriguing. While further optimization is possible, the performance of TRISO particles will ultimately be limited by the thermal decomposition of SiC which becomes rapid above ~2000 °C.

As stated previously, the optimization of conventional TRISO particles is primarily the responsibility of the AGR fuel program. The related workscope included in this program will be mainly liaison with the AGR program.

2.4.1.2 New Fuel Kernels

Thermodynamically, the UCO kernel with a conventional TRISO coating system or with an advanced coating system, such as ZrC-based TRIZO, should offer superior performance compared to UO_2 by suppressing CO formation even at high burnups. The UO_2^* particle variant with a ZrC overcoating on a standard UO_2 kernel can be considered an "advanced" kernel with a conventional TRISO coating system. If the superior performance of UO_2^* observed in the previous tests (e.g., 100% Ag retention at 1500 °C for 10,000 hr) is confirmed in larger tests, it would be a very attractive candidate because it is easy to manufacture.

Other more exotic kernel designs could be considered as well. For example, nitride kernels in coated particles are currently be investigated for possible use in gas-cooled fast reactors; however, nitride kernels require the use of enriched N-15 to prevent the formation of large quantities of C-14 via a (n,p) reaction. Nevertheless, the progress of this development effort will be closely monitored for possible application in thermal-spectrum VHTRs if performance advantages are demonstrated.

A fundamental problem with all candidate kernel materials, including nitrides as well as conventional carbide and oxide kernels, is that fission gases, including radiologically important iodine isotopes, the volatile fission metals (Ag, Cs, etc.) and the noble metals, especially Pd isotopes, are not likely to be chemically bound in refractory compounds in the kernel at reactor operating temperatures.

2.4.1.3 New Coating Systems

New particle coating systems could include the use of more refractory ceramic materials to replace the SiC and/or the PyC coatings (e.g., ZrC replacing the SiC) or the use of several proven coatings in a single particle (e.g., a ZrC coating interior to the SiC coating in combination with PyC coatings). A broad spectrum of refractory ceramic coatings, including carbides, nitrides, oxides, etc., can be applied by chemical vapor deposition (CVD); in principle, many of these ceramics could be applied as coatings on fuel particles by CVD in fluidized beds (i.e., by conventional particle coating technology). The practical problem with all exotic new coating systems is that extensive R&D would be required, including process development, irradiation tests, and postirradiation heating tests, before relevant data could be obtained to determine whether they offered any significant performance advantages. The impact on the nuclear design would also have to be assessed (e.g., materials with high absorption cross sections are not attractive).

2.4.1.3.1 Refractory Barrier Coatings

There are metal carbides that are more refractory than SiC. The highest melting point materials are hafnium carbide (HfC), tantalum carbide (TaC), zirconium carbide (ZrC), and niobium carbide (NbC); selected material properties of these refractory carbides are compared with SiC and C in Table 2-1. All of these materials can be formed by CVD at temperatures about 1/3 of their melting points. Based upon open literature publications, only ZrC has been seriously investigated as a replacement for SiC in coated fuel particles; however, as discussed in Section 3, classified programs involving nuclear rocket propulsion systems with coated-particle fuels may have investigated other carbide coatings as well.

These refractory carbides do have higher melting points and much higher densities than SiC (the latter may imply that they would be a more effective diffusion barrier to volatile fission metals, such as silver). However, their effectiveness as a metal release barrier can not be predicted with confidence because transport mechanisms for volatile fission metals in ceramic coatings are not well established (even for SiC). Moreover, their mechanical/structural stability as a particle coating under neutron irradiation would have to be determined by extensive irradiation testing. Hafnium, which is widely used a control material in power reactors, has a large neutron absorption cross section which probably eliminates HfC from consideration; the cross section for Ta is also significant. Perhaps the Achilles heal of all these refractory carbides is that they oxidize in air at ~800 ^{o}C. The essential issue may be whether they undergo passive oxidation like SiC or active oxidation like ZrC. To elaborate, when SiC is oxidized in air at elevated temperature, it forms a coherent SiO_2 layer which serves as a diffusion barrier to oxygen, thereby limiting the rate of further oxidation; in contrast, ZrC does not form a protective oxide layer, and the rate of oxidation is determined by the rate of bulk mass transfer of the oxidant to the surface. The safety implications for unlimited air ingress accidents need to be determined.

2.4.1.3.2 Multiple Diverse Barrier Coatings

An alternative approach would be to use a particle that contained multiple diverse barrier coatings. One combination that appears attractive would be a coating system that contains a ZrC layer interior to a SiC layer (e.g., PyC/ZrC/PyC/SiC/PyC); in fact, the UO_2^* particle with a ZrC overcoat on the kernel is a variant of this strategy. In principle, the ZrC layer would getter excess oxygen, thereby suppressing CO formation, and it might serve as a diffusion barrier to Pd isotopes, thereby preventing them from attacking the SiC (although Pd may diffuse through ZrC without reacting with it); in turn, the SiC layer would protect the ZrC layer from external oxidants. The disadvantages of such a particle are likely greater manufacturing costs (more coating steps and QC measurements) and lower heavy-metal loadings because of a larger coating thickness-to-kernel diameter ratio.

2.4.1.4 Getters

Instead of employing more refractory barrier coatings, an alternative strategy for providing a higher temperature capability for coated fuel particles is the use of getters to chemically bind excess oxygen liberated upon fission to suppress CO formation and/or to bind corrosive agents, such as Pd, produced by fission. These getters could be present as kernel additives, distinct coating layers, or co-deposited in other layers. As described above, the carbide phase in a UCO kernel is in effect a kernel getter to bind excess oxygen. The ZrC phase in a UO_2^* particle, which is also assumed to function as an oxygen getter, may be present as a distinct layer (e.g., a ZrC overcoat on the kernel) or co-deposited in the buffer layer.

Since Pd isotope react with the SiC coating in conventional TRISO particles, forming palladium silicides, one approach for mitigating this effect is provide sacrificial Si interior to the structural SiC layer to getter the Pd. This SiC could be included as a kernel additive or could be in the form of a sacrificial SiC layer. Another possibility would be to replace the IPyC layer with a Si-BISO layer (a layer of co-deposited PyC and SiC).

The Japanese have investigated the addition of sacrificial SiC to the particle to protect the main structural SiC layer (Minato 1997). Three different variants to a conventional TRISO-coated UO_2 particle were fabricated and tested: (1) an additional layer of SiC+PyC ("Si-BISO") adjacent to the interior of the structural SiC layer; (2) a layer SiC+PyC separated from the structural SiC layer by a dense PyC layer; and (3) a pure sacrificial SiC layer separated from the structural SiC layer by a dense PyC layer. In all cases, the advanced coating systems had good irradiation performance, and the additional layers of SiC and SiC+PyC trapped palladium effectively to prevent the corrosion of the structural SiC layer.

2.4.2 Improved Fuel Element Designs

As previously mentioned, core design changes have been identified that should permit increased core outlet temperatures without a proportionate increase in peak fuel temperatures during normal operation (e.g., MacDonald 2003b). While beyond the scope of this advanced fuel development plan, there will be considerable synergy between these design efforts and the development of advanced fuels. If fuel-element and core design changes can accommodate higher core outlet temperatures without a proportionate increase in peak fuel temperatures, then the development of advanced fuel with higher temperature capabilities would provide the design options of higher core power densities (i.e., higher core thermal power) or even higher core outlet temperatures.

Fuel-element design changes under consideration for the VHTR include the optimization of the reference prismatic fuel element (e.g., an 11-row block instead of the standard 10-row block to reduce linear heat rates) and the use of nonconventional fuel element designs (e.g., molded block, direct cooling, etc.). Section 7 of IAEA-TECDOC-978 reviews several advanced fuel element designs, including a molded block (to eliminate the gap between the fuel compact and the graphite block). Core design changes under consideration for the VHTR include the introduction of fixed column orifices to improve the coolant flow distribution and various fuel shuffling schemes to further optimize the power distribution (MacDonald 2003b).

2.5 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 (Section 4). 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[5] with a 1000 °C core outlet temperature.
- 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 advanced fuel 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).

2.6 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 UCO 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).

---

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

2.7 Plan Organization and Content

This fuel development 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 background. It briefly describes the VHTR concept, the challenges and development goals for VHTR applications of coated-particle fuels, and the strategy for the development of high temperature, coated particle fuel.

Section 3 describes the knowledge base for fabrication of coated particles, the experience with irradiation performance of coated particle fuels, and the database for fission product transport in HTGR cores.

Section 4 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 4 also presents limits on the irradiation conditions to which the coated particle fuel can be subjected for the core design. These limits are based on past irradiation experience.

Section 5 describes the Design Data Needs to: (1) fabricate the coated particle fuel, (2) predict its performance in the reactor core, and (3) predict the radionuclide transport throughout the VHTR plant.

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

Section 7 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 8 states the Quality Assurance requirements that will be applied to the development activities.

Section 9 presents cost and schedule estimates, organized by a simple Work Breakdown Structure (WBS).

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

2.8 References for Section 2

[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.

Bullock, R. E., and C. A. Young, “TRISO Fuel Particle Coating Design Basis,” DOE-GT-MHR-100225, Rev. 0, General Atomics, September 1994.

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

Hanson, D. L., "Screening Tests for the Selection of VHTR Advanced Fuel," PC-000510, Rev. 0, General Atomics, December 2003.

Ho, F., "Materials Models of Pyrocarbon and Pyrolytic Silicon Carbide," CEGA-002820, Rev. 1, General Atomics, July 1993.

Homan, F. J., et al., "Stoichiometric Effects on Performance of High-Temperature Gas-Cooled Reactor Fuels from the U-C-O System," Nucl. Tech., **35**, 1977, pp. 428-441.

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

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.

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.

Minato, K., et al., "Advanced Coatings for HTGR Fuel Particles against Corrosion of SiC Layer," J. Nucl. Mat., **246**, 1997, pp. 215-222.

Myers, B. F., "Fuel Design Data Manual," GA Document 901866, Rev. F, General Atomics, August 1987.

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

[PSID] "Preliminary Safety Information Document for the Standard MHTGR," Volumes 1-6, HTGR-84-024, Amendment B, August, 1992.

Petti, D. A., et al.., "Key Differences in the Fabrication, Irradiation and Safety Testing of U.S. and German TRISO-coated Particle Fuel and Their Implications on Fuel Performance," INEEL/EXT-02-00300, INEEL, June 2002.

[RF Plan] "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," GA Report FDO-EDO-N-TRT-X-000132, General Atomics, December, 2001.

Table 2-1. Properties of Refractory Carbides

PROPERTY/MATERIAL[6]	C	SIC	HFC	TAC	NBC	ZRC
Molecular Weight (g/mol)	12.01	40.10	190.54	192.96	104.92	103.23
Density (g/cc)	2.25	3.21	12.70	14.50	7.79	6.59
Thermal absorption cross section (b)	0.0036	0.16	104	21.5	1.13	0.18
Resonance integral (b)	0.0016	0.08	2000	720	8.6	1.0
Melting Point (ºC)	---	2827[7]	3890	3880	3500	3540
Boiling Point ((ºC)	3550	N/A	5400	5500	N/A	5100
Thermal Expansion (ppm/ºC)	10.0	5.8	6.8	6.6	6.9	7.3
Thermal Conductivity (W/m-ºC)	150	50	22	22	30	20
Specific Heat (J/g-ºC)	0.84	0.58	0.20	0.19	0.35	0.37
Enthalpy (kJ/H^o_{Tm})	8.58	---	1.11	1.11	1.86	1.79
Hardness (kg/mm2)	20	2800	2300	2500	2400	2700
Crystal Structure[8]	HEX	FCC	FCC	FCC	FCC	FCC
Transverse Rupture Strength (MPa)		588		350-400	300-400	---
Young's Modulus (GPa)		466	350-510	285-560	338-580	350-440
Poisson's Ratio		0.21	0.18	0.24	0.21	0.191
Oxidizes in air @ T (ºC)	500	~800[9]	500	800	800	800

---

[6] Property data from http://www.ultramet.com/old/carbide.htm; the data given for SiC are not necessarily applicable to SiC coatings in TRISO fuel particles.

[7] Sublimes, decomposes into elemental constituents.

[8] HEX = hexagonal; FCC = face centered cubic

[9] Typical value based upon measurements on TRISO particles; reference http://www.ultramet.com/old/carbide.htm does not provide quantitative definition of "oxidizes in air."

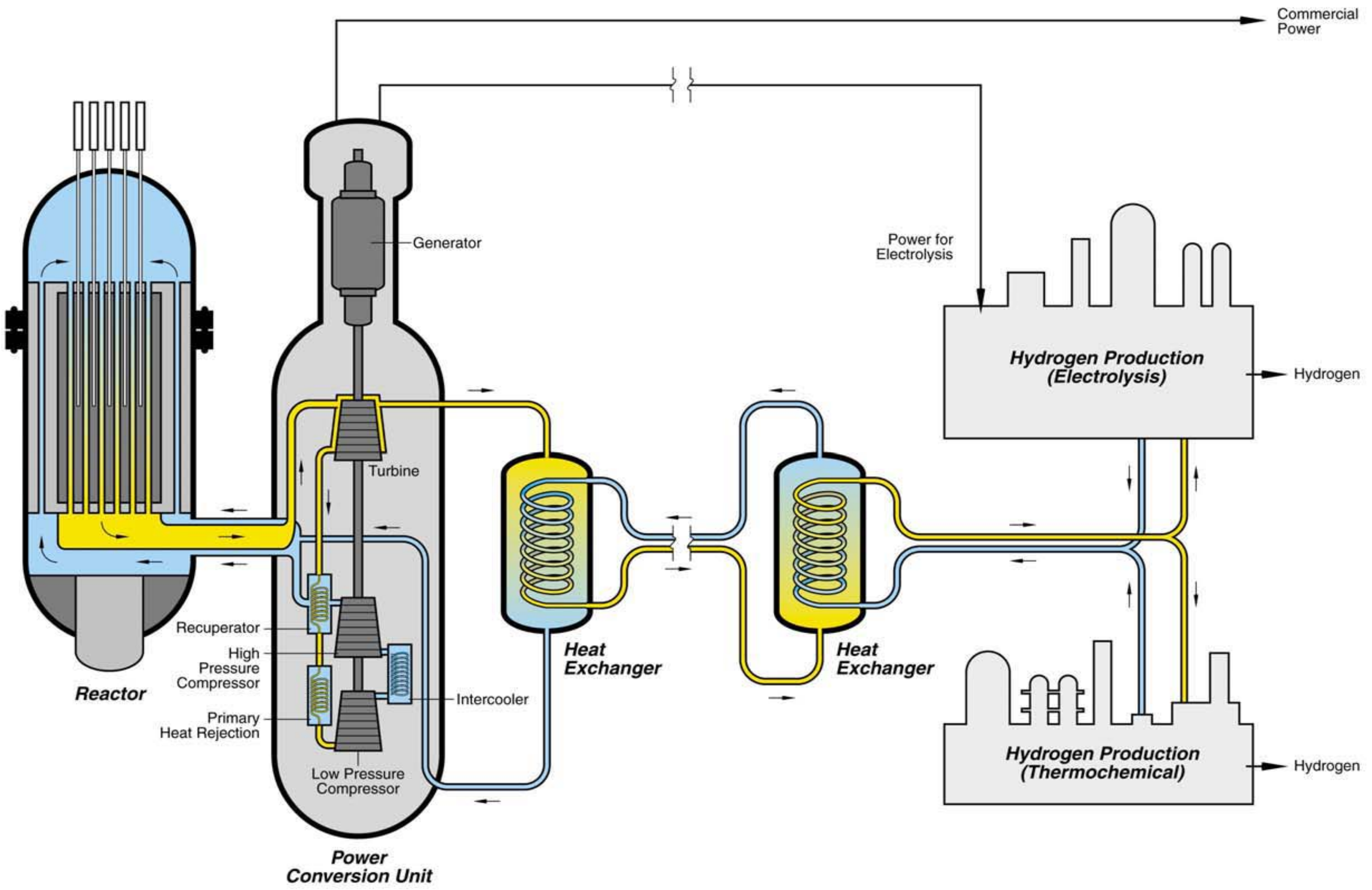


Figure 2-1. VHTR Demonstration Module

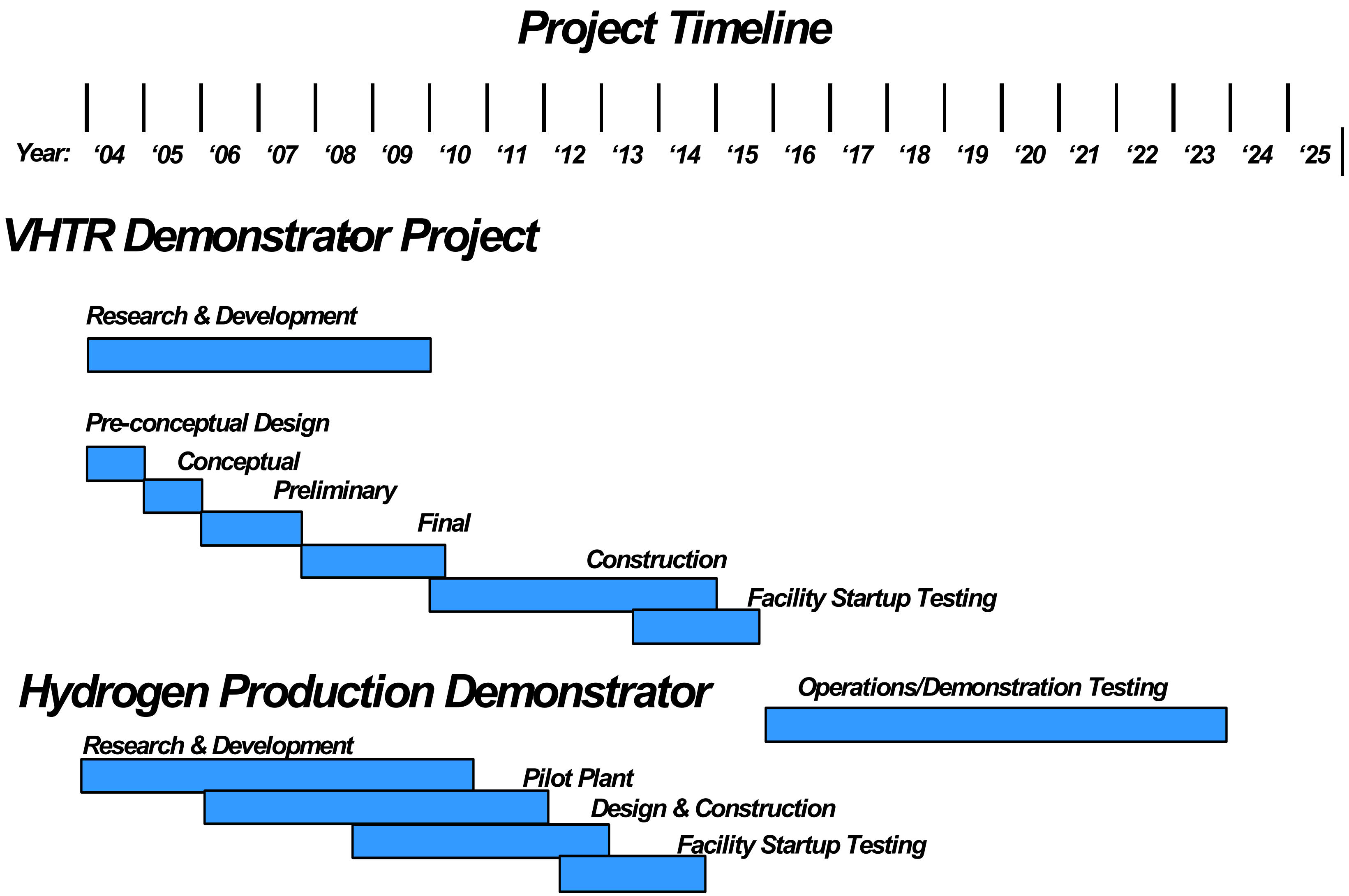
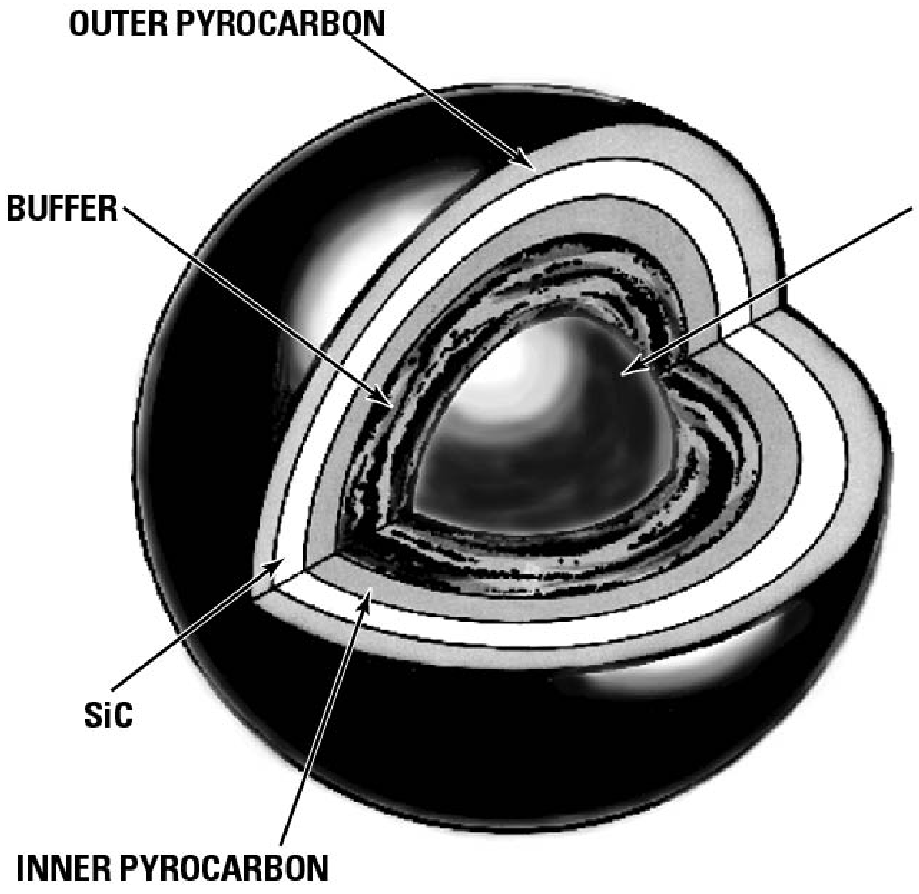


Figure 2-2. VHTR Demonstration Module Master Schedule



COMPONENT/PURPOSE

- **FUEL KERNEL**
  - – PROVIDE FISSION ENERGY
  - – RETAIN SHORT LIVED FISSION PRODUCTS

- **BUFFER LAYER (POROUS CARBON LAYER)**
  - – ATTENUATE FISSION RECOILS
  - – VOID VOLUME FOR FISSION GASES

- **INNER PYROCARBON (IPyC)**
  - – PROVIDE SUBSTRATE FOR SiC DURING MANUFACTURE
  - – PREVENT Cl ATTACK OF KERNEL DURING MANUFACTURE

- **SILICON CARBIDE (SiC)**
  - – PRIMARY LOAD BEARING MEMBER
  - – RETAIN GAS AND METAL FISSION PRODUCTS

- **OUTER PYROCARBON (OPyC)**
  - – PROVIDE BONDING SURFACE FOR COMPACTING
  - – PROVIDE FISSION PRODUCT BARRIER IN PARTICLES WITH DEFECTIVE SiC

Figure 2-3. TRISO Fuel Particle

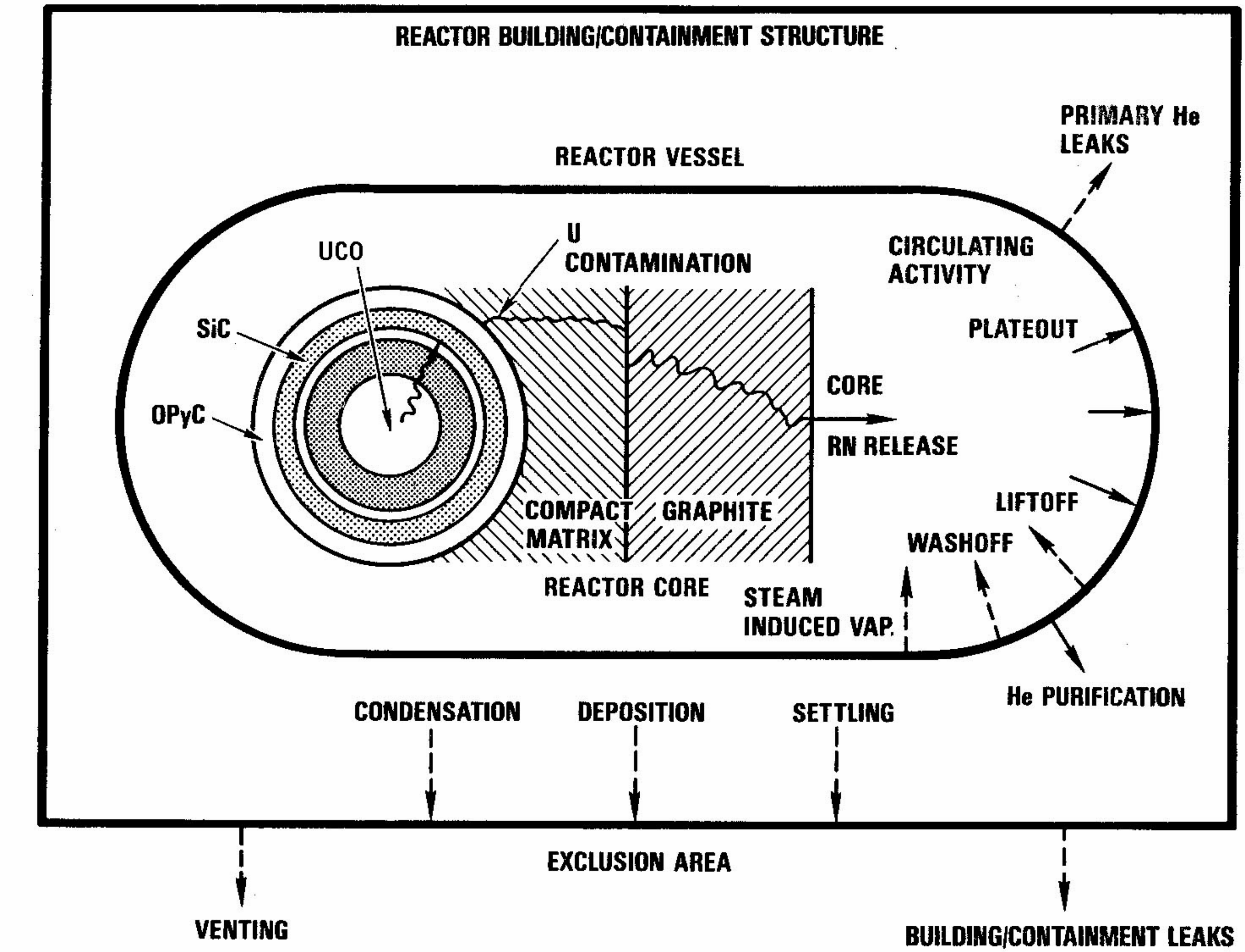


Figure 2-4. Radionuclide Containment System

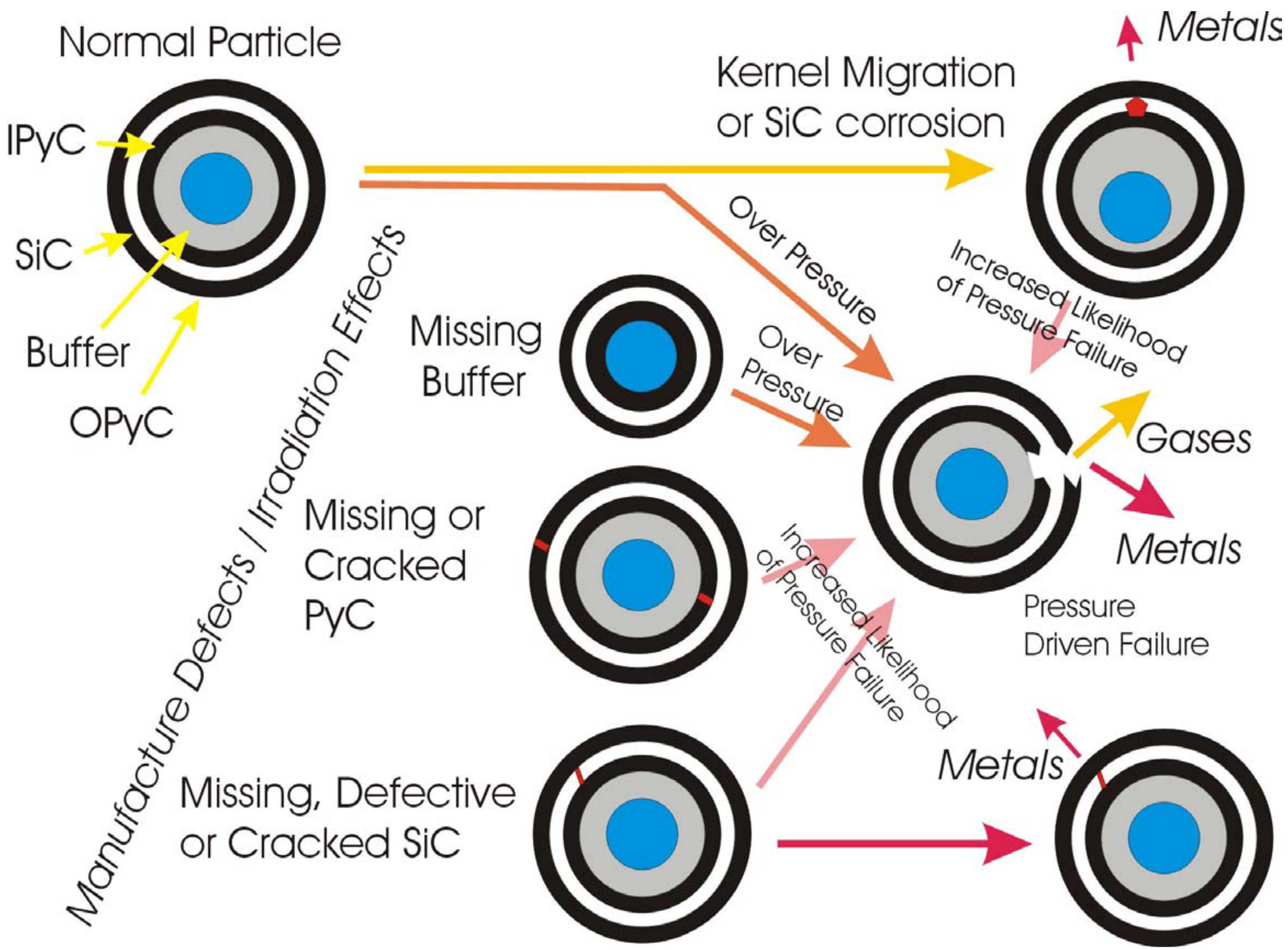


Figure 2-5. TRISO Particle Failure Mechanisms

3. Fuel Development Status

3.1 Fuel Process Development

TRISO particle fuel (with a SiC coating) has been fabricated in many countries throughout the world, irradiated in numerous irradiation test capsules, and used as the fuel in eight power and experimental reactors (IAEA-TECDOC-978 1997, IAEA-TECDOC-1198 2001). The processes for manufacturing coated-particle fuel have described previously (e.g., Bresnick 1991, Pfahls 1991). The capability of TRISO fuel particles to meet MHR performance requirements has been demonstrated in Germany for the pebble bed reactor design but has not yet been demonstrated in the U.S. for prismatic core designs.

The Advanced Gas Reactor Fuel Development and Qualification Program (AGR Program) is currently working to develop and qualify fuel for the VHTR by adapting the German coating technology to the VHTR particle design. As the starting point for the VHTR fuel development and qualification effort, the AGR Program has selected SiC TRISO-coated UCO as the reference fuel particle design. According to the AGR Plan (ORNL/TM-2002/262 2003), "it is expected that this fuel will permit high-burnup operation (22% FIMA) at fuel temperatures up to 1250°C with fast neutron fluences to 4×10^{25} n/m^2, supporting approach to the lower range of VHTR core outlet temperatures (850°C - 1000°C); but new coatings, such as ZrC, may be required for higher core outlet temperatures."

Fuel particles with ZrC and graded ZrC-PyC coatings have been in development as a potential replacement for TRISO fuel in VHTRs since the early 1970's and have been tested in several irradiation test capsules and accident simulation tests with mixed results (Section 3.2.2.2). However, systematic process development of ZrC, supported by irradiation and safety testing, to define the performance potential for ZrC coatings has not been conducted. Clearly, further research and development is required to determine the true potential of ZrC-coated fuel as an advanced HTGR fuel.

This process status section focuses upon coated particle designs with SiC- and ZrC-based coating systems since little or no information was found in the open literature regarding processes for depositing more exotic ceramic coatings (e.g., NbC, etc.) on nuclear fuel kernels. There is an extensive open literature regarding the application of a broad spectrum of ceramic coatings on various substrates by chemical vapor deposition processes which is not reviewed here. This literature would presumably be the point of departure for developing processes for applying such coatings on fuel kernels by CVD in conventional fluidized bed coaters.

3.1.1 Kernel Fabrication

Ammonia-based precipitation processes, with the two most frequently used variants referred to as "internal gelation" (e.g., Collins 1987) and "external gelation," (e.g., Naefe 1979) have rendered the traditional sol-gel processes obsolete. Both internal and external gelation processes use a concentrated heavy-metal nitrate feed that is "acid-deficient", and therefore require a concentration/denitration solution preparation step. These newer gel-precipitation methods are simpler, more reliable, and more suitable for scale-up and adaptation to radiochemical processing than the traditional sol-gel techniques.

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

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

Several researchers (e.g., Homan 1977) have investigated the benefits of including oxygen getters in UO_2 kernels to reduce the oxygen potential of the kernels, thereby reducing the potential for kernel migration (i.e., the "amoeba effect") or of pressure vessel failure of the coated particle due to excessive CO buildup. Indeed, the UCO kernel, which is actually a finely dispersed mixture of UO_2 and UC_2, is an example of a gettered UO_2 kernel. The addition of other oxygen getters to UO_2 kernels, including SiC, ZrC, Ce_2O_3 and La_2O_3, has also been investigated. All of these getters have been determined to be effective in reducing the CO pressure in irradiated UO_2 particles, with the most effective getters being UC_2, SiC, and ZrC (Proksch 1986).

General Atomics also investigated a different approach to oxygen gettering of UO_2 kernels in which ZrC was put outside the kernel, as opposed to incorporating it in the kernel (Bullock 1983). In this approach, ZrC was deposited as a thin layer on the kernel (after application of a thin PyC seal coat to protect the kernel from chlorine attack during ZrC coating), and a standard TRISO coating was then applied to the ZrC-coated UO_2 kernel. In a variation of this particle design, ZrC was dispersed in the buffer layer instead of being deposited as a ZrC coating layer. These particle designs are identified in this plan as UO_2*-C and UO_2*-B, respectively. These particles exhibited excellent irradiation performance with a complete absence of kernel migration, indicating that this is also an effective approach to reducing the oxygen potential of UO_2 kernels.

3.1.2 Coated Particle Fabrication

The advanced particle designs selected for development include ZrC, either as an addition to, or a substitute for the SiC coating layer in the standard TRISO particle design. The basic principles for chemical vapor deposition of high-quality ZrC coatings are not nearly as well known as the principles for SiC deposition, which have been developed and refined by nearly four decades of international research and development. Several researchers have deposited coating layers of

ZrC or ZrC-C alloy on microspheres by chemical vapor deposition using processes involving the reaction of a zirconium halide with a hydrocarbon gas. All of these ZrC coating processes have employed a gas mixture of Zr-halide, CH_4 or C_3H_6, H_2, and Ar.

Two approaches have been used to deliver Zr-halide to the coating chamber. One approach involves reacting a halide with hot zirconium sponge in an external Zr-halide reactor and sweeping the resultant vapor into the coater. Chlorine, bromine, methyl iodide, and methlylene dichloride have all been reacted with Zr sponge to generate Zr-halide vapor in this fashion. JAERI has fabricated ZrC coatings using all of these halides and has concluded that the bromide process is the most convenient and reliable (Ogawa 1981). In the other approach, $ZrCl_4$ powder (which is a solid at room temperature and sublimes at 352 °C) was added to the gas stream and sublimed prior to entry into the coater. Although $ZrCl_4$ powder is highly hydroscopic and is difficult to meter precisely, LASL (now LANL) was successful in developing a motor driven feeder that precisely and reproducibly fed $ZrCl_4$ powder into the coating chamber (Hollabaugh 1977). LASL specifically developed the $ZrCl_4$ powder feeder in response to difficulties encountered by early ZrC coating researchers in quantitative control of the flow rate of Zr-halide by either the $ZrCl_4$ sublimation approach or the reaction of Zr sponge with halide approach.

Systematic ZrC coating studies to determine the influence of the concentrations of Zr-halide, CH_4 or C_3H_6, and H_2 on the ZrC deposition rate and on the nature of the ZrC deposit have been performed by both JAERI (Ogawa 1981) and LASL (Hollabaugh 1977). Both studies determined that the methane (or propylene) concentration is the rate controlling factor for ZrC deposition and that the CH_4/Zr-halide ratio and hydrogen concentration can be adjusted to obtain the desired stoichiometry (i.e., C/Zr ratio) of the deposit. If the CH_4/Zr-halide concentration ratio is too high, free carbon will be co-deposited with ZrC, which is undesirable because ZrC-C alloy coatings have been found to be nonretentive of metallic fission products. Increasing the H_2 concentration inhibits the pyrolysis of methane, which permits ZrC to be deposited at higher methane concentrations (i.e., at a higher deposition rate) without obtaining free carbon in the ZrC layer. Practical coating rates for ZrC are from about 0.2 μm/min to 0.5 μm/min, which is about the same as for SiC. The results of these coating studies were consistent with thermochemical predictions, indicating a reasonably good understanding of the process.

The JAERI study also determined the effect of coating temperature on the ZrC deposition rate and the nature of the PyC deposit. Methane pyrolysis efficiency (and therefore the ZrC deposition rate) is at a maximum at about 1500°C and decreases rapidly with decreasing or increasing temperature. At temperatures above about 1550°C, the ZrC coatings had a blistered surface morphology, which corresponded with a decrease in density from 6.6 g/cc at 1500°C to 6.1 g/cc at 1550°C. The optimum coating temperature would therefore appear to be about 1500°C, which is also approximately the optimum coating temperature for SiC.

Based on the above, the process conditions (for a laboratory-scale spouting bed coater) and properties of the resultant ZrC coatings should be as follow

Process

- Coating temperature: 1500°C
- Source of Zr: In-situ generation of $ZrBr_4$ or sublimation of $ZrCl_4$
- Source of C: CH_4

- Fluidization gases: Ar and H_2 µm/min
- Coating Gas C/Zr (mole fraction): ≈1
- Mole fraction H_2: ≈0.25
- Linear coating rate: ≈0.3 – 0.5

Desired Properties of ZrC

- Density: ≥6.6 g/cc
- C/Zr: ~1.0 (no free carbon)
- Surface Appearance: Smooth and shiny

Whereas the relationship between SiC microstructure and irradiation performance is reasonably well known, the limited research and development with ZrC has not been sufficient to establish a good correlation between ZrC microstructural properties and irradiation performance and fission product retentiveness. However, in general, the microstructure of the ZrC as revealed in a metallographic cross section should be smooth and uniform without evidence of porosity or structural banding.

It is important to note that, to date, all ZrC coating has been done in laboratory-scale coaters, and that accurate and reliable feeding of Zr-halide into the coater is likely to be far more difficult in a production-scale coater than in a laboratory-scale coater because of the much larger quantities of Zr-halide required. This may represent a considerable scale-up challenge for ZrC-TRISO fuel production that does not exist for standard TRISO fuel particle production.

3.1.3 Compact Fabrication

Extensive experience in the production of cylindrical fuel compacts was gained at GA during production of Fort St. Vrain (FSV) fuel (McEachern 2001b). In the FSV compacting process, a thermoplastic matrix composed primarily of graphite powder and petroleum pitch (as the binder) was injected into a bed of particles in a mold to form a compact. This process was developed and used for FSV fuel production primarily because of its suitability for making compacts with high particle packing fractions. However, this process has a number of drawbacks. The injection process requires compaction of the bed of particles wherein adjacent particles are in direct contact, which is a potential source of coating breakage. Also, the compacts must be supported by alumina powder during carbonization to prevent them from losing their shape. Furthermore the petroleum pitch and alumina powder are sources of impurities that are known to attack SiC coatings.

The fuel quality requirements for current MHR designs are much more stringent than for FSV so the compact fabrication process must be capable of reducing the level of heavy-metal contamination and defective particles in compacts by more than an order of magnitude compared to the levels demonstrated during FSV fuel production. To achieve this capability, a compacting process improvement program was conducted at General Atomics in 1995-1996 (Goodin 1996). High-quality, German LEU UO_2 particles were used in the study to demonstrate the fabrication of fuel compacts meeting commercial GT-MHR specifications for as-manufactured heavy-metal contamination and coating defects in the finished compacts. Improvements were made in the control of the bed compacting and matrix-injection steps, which significantly reduced the mechanical stresses in the coated particles during the formation of green compacts.

More importantly, these process studies confirmed the results of earlier studies: that a major source of heavy-metal contamination and SiC coating defects in fuel compacts was due to SiC chemical reactions with impurities, especially iron and other transition metals, from the petroleum-pitch matrix and introduced into the compacts in the high-temperature heat treatment furnace. After process changes were made to reduce impurity levels, fuel compacts (referred to as "MHR-1 compacts") that met commercial GT-MHR product quality specifications with large margins were fabricated.

While GA developed and utilized a thermoplastic-matrix based compacting process (which required compression of beds of particles in direct contact) because of the high fuel particle packing fraction requirements for FSV, the rest of the international HTGR community focused on thermosetting-matrix-based compacting processes. With these latter processes, the loose particles were first overcoated with a compliant matrix material which eliminated direct particle-to-particle contact during compacting; however, particle overcoating limits the particle packing fraction in the fuel compact or sphere to lower values (< ~50%) than the FSV process. Successful compacting in which a synthetic thermosetting resin was used as the binder, was demonstrated for fuel elements containing overcoated fuel particles for the pebble bed reactor programs in Germany (Nabielek 1990) and in China (Chunhe 2001). Annular fuel compacts have been developed in Japan (Saito 1991) using a similar process in which the fuel particles are overcoated with a thermosetting matrix. The French company, CERCA, studied a process where matrix flakes were mixed with particles and the compacts formed in a mold by melting the matrix. All of these compacting processes that employ overcoating are suitable for fabrication of fuel elements having relatively low (i.e., < ~50%) particle packing fractions. Work was also conducted at GA to develop an injectable thermosetting matrix process suitable for fabrication of fuel compacts having high particle packing fractions (Bullock 1976, Bullock 1977, Bullock 1980, Bullock 1981).

3.1.4 Quality Control Techniques

QC methods for TRISO fuel particles are well established and have been used for relatively large-scale fuel production in the U.S. and Germany (e.g., Bresnick 1991, Pfahls 1991). However, the fuel product specifications that have been used historically for TRISO fuel are not sufficiently comprehensive to ensure the required irradiation performance of the fuel; consequently, a combination of product specifications and process specifications have been used to ensure the necessary product quality. The AGR Plan (ORNL/TM-2002/262 2003) has identified a need to develop new QC methods for characterization of the stoichiometry of individual UCO kernels and of the IPyC coating permeability and anisotropy. Enhanced methods for characterizing SiC microstructure and defects may also be needed, but this has not been established conclusively. Furthermore, many of the existing QC methods employ 1970's technology and are too time consuming and costly to support economical large-scale fuel production. Accordingly, the AGR Plan has also identified a need to develop low-cost automated nondestructive methods capable of high throughput rates and providing near real-time feedback to the fuel fabrication processes. The AGR Plan includes work scope to satisfy the identified QC needs.

Because ZrC, in contrast to SiC, does not form a protective oxide layer when exposed to air at high temperature (and therefore completely oxidizes to ZrO_2), several of the QC methods employed for TRISO fuel cannot be used for ZrC-TRISO fuel. This includes the burn-leach test for defective SiC, and all QC methods in which coated particles or coating fragments are "burned-back" in air to segregate the SiC layer from pyrocarbon. The much higher density of ZrC (6.6 g/cm^3) relative to that of SiC (3.2 g/cm^3) is also problematic with respect to coating density measurements (i.e., unavailability of suitable heavy liquids for the density gradient column technique) and the use of x-radiographic inspection for coating thickness, missing buffer coatings, and fuel dispersion.

Some progress has been made in developing new QC methods for ZrC coatings, but this progress has been limited. Ogawa and Fukuda (Ogawa 1989) developed a plasma oxidation technique that is capable of completely removing pyrocarbon from ZrC without significant oxidation of the ZrC. Ogawa and Fukuda (Ogawa 1990) have also applied plasma oxidation with emission monitoring to the quantitative analysis of the free carbon content in ZrC coatings, a property which is very important to the fission product retentiveness of ZrC.

3.2 Fuel Materials Development

3.2.1 TRISO-Coated Fuels

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

3.2.2 Advanced Coated-Particle Fuels

3.2.2.1 UO_2 Fuel*

Both UO_2*-B (ZrC dispersed in buffer) and UO_2*-C (ZrC kernel overcoat) particles were irradiated in HRB-15A (high-temperature experiment) and HRB-15B (low-temperature experiment) and subjected to post-irradiation heating tests with the following results (Bullock 1983):

- There was no evidence of kernel migration in the UO_2* particles in HRB-15A, but migration was observed in ungettered UO_2.
- There were no failures of the SiC or OPyC coatings in the 1800 loose UO_2*-C particles irradiated in HRB-15B (however, no kernel migration or pressure vessel failure was observed in ungettered UO_2 either). Surprisingly, the ZrC layer was determined to have remained intact in all of the particles randomly selected for post-irradiation heating tests.
- Isothermal annealing tests at 1500 °C revealed great improvement in fission product retention of HRB-15B UO_2*-C particles (that unexpectedly had stayed intact throughout the irradiation) relative to standard TRISO particles.
- The ZrC layer failed in about 22% of the UO_2*-C particles in HRB-15A (~100% failure was expected during irradiation).

- UO_2*-B particles had better fission product retention than ungettered UO_2 particles, but had more fission product release than UO_2*-C particles.
- The post-irradiation heating tests provided the first ever demonstration that coated particles can completely retain highly diffusive Ag-110m at 1500 °C for times as long as 10,000 hours.

UO_2^* particles were also irradiated in capsule HRB-16 with the following results (Ketterer 1985):

- There were no observed OPyC failure in 436 particles, which implies no SiC failure.
- From the IMGA analysis none of the 640 particles recovered by deconsolidation of rod 1 were found to have released Cs-137 during irradiation to 3×10^{25} n/m^2 fast fluence at 1000 °C, and only one of 454 particles in rod 12 released Cs-137 after exposure to 6.2×10^{25} n/m^2 fast fluence at 1120 °C. In contrast, eight of 439 UCO particles released Cs during exposure of 4.3×10^{25} n/m^2 fast fluence at 1075 °C.
- Ag-110m was retained at near-100% levels in the UO_2* particles from rod 12.
- Although not directly determined, ZrC failure in HRB-16 particles should have been similar to the 22% failure rate found in HRB-15A, yet silver retention from particle to particle was very consistent, suggesting that an intact ZrC layer throughout irradiation is not necessary to achieve improved Ag retention. This suggests that an improved SiC layer made possible by the initial presence of an intact ZrC layer may be responsible for the better retention of silver in UO_2* particles.

Based on the results from the HRB-15A, HRB-15B, and HRB-16 capsule irradiations and on the post-irradiation heating tests, it is concluded that UO_2* fuel particles can potentially provide significantly improved fission product retention relative to standard TRISO particles due to both reduced SiC defect fractions and better retention of radionuclides in particles with intact SiC coatings. Furthermore, if the ZrC remains intact in UO_2*-C particles (as it did 78% of the time in HRB-15A and 100% of the time in HRB-15B), the ZrC layer serves as an additional diffusion barrier to fission products and reduces the swelling of UO_2 that occurs in unconstrained kernels. The reduction in kernel swelling reduces the porosity in the kernel at the end of irradiation, which results in increased retention of fission products by the kernel itself.

Therefore, the improved fission product retentiveness of UO_2*-C fuel particles relative to standard TRISO fuel particles may be attributed to three factors:

- Improved SiC coating quality (due to sealing of the kernel by ZrC prior to SiC deposition),
- Fission product gettering by the ZrC layer itself, and
- Better fission product retention by kernels that are prevented from swelling by the intact ZrC layer.

The UO_2*-B particle design benefits from only the second of these factors, a consideration which would appear to make UO_2*-C a better design than UO_2*-B.

3.2.2.2 TRIZO Fuel

The testing done to date on coated fuel particles with ZrC coatings is very limited compared with the testing performed on standard TRISO particles. Much of the data, particularly in the U.S., have been collected on isolated specimens included in irradiation tests whose primary purpose was to test reference fuel designs. A number of summaries of the database for TRIZO fuel have been performed. (Homan 1985) reviews all of the irradiation test data collected within the U.S. program on HTGR fuel particles with ZrC coatings, including UO_2*. (Kasten et al. 1991) provides a summary of the international fabrication, irradiation, and post-irradiation heating experience. Chapter 7 of (IAEA-TECDOC-978, 1997) also provides a review of the international experience with TRIZO. The following summary attempts to highlight the key data and conclusions rather than providing an exhaustive review of all of the available data. The reader is referred to the above references for a more extensive review of the ZrC database.

In the U.S., TRIZO particles fabricated at LASL and at GA were irradiated in capsules HT-31, HT-32, HRB-7, HRB-8, HRB-12, and HRB-15a. The results of these tests are summarized in (Homan 1985), which concludes that the fission product retention of particles with ZrC coatings irradiated in these tests was generally inferior to that of similar particles with the TRISO design primarily due to relatively high in-pile failure fractions for the TRIZO particles. However, the irradiation test results for these capsules also show that ZrC is, as expected, more resistant to chemical attack by fission products than is SiC. Furthermore, Homan emphasizes that the fabrication process for ZrC coatings had not been optimized prior to fabrication of the various TRIZO samples for these irradiation tests.

In Japan, ZrC-coated fuel particles were under development at JAERI for more than 20 years as an advanced HTGR fuel for very high temperature application. This effort culminated in development of the bromide process for ZrC deposition and in a number of irradiation tests and post-irradiation heating tests on lightly irradiated particles in which the particles performed much better than in the earlier U.S. tests (Ogawa 1992), Kr-88 R/B measurements for the 78F-4A irradiation test in the JMTR indicated that there was no TRIZO coating failure during irradiation at about 900°C. Acid leaching of TRIZO particles from deconsolidated compacts irradiated in capsules 78F-4A and 80F-4A also indicated no irradiation-induced failure. Metallographic examination of TRIZO particles from these capsule tests and from capsule ICF26H (including a particle irradiated at about 1600°C revealed no evidence of ZrC coating deterioration due to fission product attack. TRIZO particles irradiated in 78F-4A were subjected to a postirradiation heating test in which the particles were kept at 2400°C for about 6000 seconds; there was only one failure among the 101 particles tested. This test revealed that ZrC can sustain very large strains at this temperature.

In a 1996 review of HTTR fuel production technology, (Hayashi et al 1996) compared ZrC with SiC as follows based on the results from the JAERI TRIZO development program up to that time:

- Better performance against kernel migration failure and chemical attack by fission products, including Pd,
- Better durability against pressure vessel failure at temperatures above 1600 °C (up to at least 2000 °C),
- Better retention of fission products, except Ru, particularly at temperatures of 1600°C and above.

Specifically, in heating tests from 1400°C to 2400°C, the fractional release of Cs-137 was much smaller for ZrC-coated particles than SiC-coated particles. However, the fractional release of 365-day Ru-106 was much larger for ZrC-coated particles at 2000°C to 2300°C. This result is consistent with the results at heating temperatures of 1400°C to 1600°C obtained by other researchers. Hayashi et al concluded that the results of the ZrC R&D at JAERI have revealed overall excellent performance of ZrC-coated fuel particles, although they noted that the high diffusivity of Ru-106 in ZrC may limit the feasibility of utilization of ZrC in gas-turbine HTGRs. They also concluded that the R&D work is limited, and that further research on an engineering scale is required in order to determine the commercial feasibility of ZrC-coated fuel as an advanced HTGR fuel; key issues include:

- Producibility of ZrC in scaled-up facilities,
- Property changes and performance during irradiation at higher temperatures to higher burnups and fast neutron fluences, and
- Safety performance under accident conditions (i.e., ZrC oxidation in water or air ingress accidents)

Further evidence of the potential superiority of ZrC to SiC for use in the VHTR was provided in a later test in the JMTR in which a single fuel compact containing TRIZO particles was irradiated along with a companion compact containing standard TRISO particles (Minato 1999). These compacts were irradiated to 4.5% FIMA at temperatures from 1400°C to 1650°C, with a time-average peak fuel temperature of about 1530°C. The through-failure fraction of the TRIZO particles irradiated in this test was 1.4×10^{-4} (which equates to zero or one particle out of the 2,400 particles in the compact). The failure fraction in the TRISO particles was 8.6×10^{-3}. This test also provided further confirmation that ZrC is much more resistant to palladium attack than is SiC. However, an interesting result of this test is that no buildup of palladium was detected at the inner surface of the ZrC layer, which suggests that ZrC coatings are not retentive of palladium. This result led Minato et al. to suggest that should ZrC prove to be nonretentive of palladium, a SiC layer inside or outside the ZrC layer might be needed in TRIZO particles in order to retain palladium.

3.2.2.3 Other Advanced Fuels

JAERI (Minato 1997) investigated the addition of three combinations of SiC and SiC + PyC (i.e., SiC with free carbon) coating layers to the standard TRISO particle design as a means of preventing corrosion of the SiC layer by fission products. These particle designs included:

- Type-A: Included a SiC + PyC layer between the IPyC and SiC layers
- Type-B: Same as Type-A but also included a dense PyC layer between the SiC and the SiC + PyC layer
- Type-C: Same as Type B except that a SiC layer was substituted for the SiC + PyC layer

In all cases the concept was to use additional SiC inside the primary SiC layer to trap fission products, thereby preventing them from reaching and attacking the primary SiC layer. In all of these particle designs, the coating thicknesses of the standard SiC and dense PyC coating layers were reduced so that the total thickness of the coating layers remained the same as in the standard TRISO particle design.

Minato reports that each of these advanced particle designs was irradiated together with standard TRISO particles in the Japan Research Reactor-2 (JRR-2) with excellent results. The burnups ranged from 3.7 to 7.0% FIMA at an irradiation temperature of 1330 °C. The advanced coatings exhibited good mechanical stability and the additional layers effectively trapped palladium to prevent corrosion of the primary SiC layer. The intermediate dense PyC layer between the SiC + PyC layer and the SiC layer in the Type-B particles and between the extra SiC layer and the primary SiC layer in the Type-C particles was found to interrupt radial extension of the corrosion zone from the additional SiC layer to the primary SiC layer.

Other more exotic advanced fuels have been, or are being investigated for use in various reactor designs. These include, for example, new fuels for fast gas-reactors in which pellets of fuel are coated with a material such as titanium nitride as an alternative to graphite.[10] Also, CVD niobium carbide-coated uranium oxide fuel and binary carbide fuels of (U, Zr)C were investigated in the KIWI and NERVA nuclear rocket propulsion programs in the 1960's.[11] Uranium bearing, solid-solution tri-carbide fuels such as (U, Zr, X)C, where X = Nb, Ta, Hf, or W are currently under development at the Innovative Nuclear Space Power and Propulsion Institute at the University of Florida for advanced space power and propulsion applications.[12] According to this reference, the presence of non-uranium carbides in the tri-carbide fuel allows for gradient coating of fuel pellets with refractory metal carbides for fission product containment, and no additional coating is necessary as with earlier graphite matrix and composite fuels. Although these advanced fuels are intriguing, it should be noted that there is nothing in the literature concerning any research and development that has been aimed toward the use of these fuels in a VHTR.

Considerable research has been done and continues in the development of inert matrix fuels (IMF) to facilitate the burning of weapons-grade plutonium and commercial-reactor plutonium (and higher actinides) in LWRs. If this burning of fissile actinides is to be accomplished in LWRs without the inclusion of uranium (for non-proliferation considerations, etc.), then inert materials which act as diluents must be added to reduce the fission rate density and the effective burnup. A broad spectrum of ceramic materials has been evaluated for use (e.g., Journal of Nuclear Materials 1999) diluents in IMF in LWRs, and some may have application in coated-particles fuels as well. Although TRISO-coated particles with highly enriched uranium and highly enriched plutonium have been successfully irradiated to high burnup (>70% FIMA), the International GT-MHR program will evaluate a TRISO-coated PuO_x kernel diluted with carbon or zirconium as a backup to the reference, TRISO-coated, pure fissile PuO_x particle.

3.2.3 Coated-Particle Performance Modeling

The coated-particle fuels that will be used for VHTR applications must satisfy the performance requirements given in Section 4, which are specified on a core-average basis. Fuel performance models in core-survey codes (Section 3.3.1) are used to predict the in-reactor performance of the coated particles under normal operating conditions and postulated accident conditions. The currently available performance models (e.g., Myers 1987) are primarily for standard kernels (UC_2, UO_2, UCO) with conventional TRISO coatings. Initially, these existing models will be

---

[10] http://www.cea.fr/gb/publications/Clefs45/clefs45gb/clefs4521a.html

[11] http://www.fas.org/nuke/space/c04rover.htm

[12] http://www.inspi.ufl.edu/tricarbide.pdf

used for candidate advanced fuel particles with modified material properties (e.g., strength) as available; however, design-specific performance models for the candidate advanced fuels will need to be as soon as the requisite experimental data become available.

In general, fuel performance is expected to vary significantly with spatial location in the reactor core, because of the expected, large spatial and time variations of temperature, burnup, and fast fluence, all of which strongly affect fuel performance. The following strategy (DB-MHR Plan 2002) is used to define a fuel particle design that is expected to satisfy performance requirements.

Existing performance models are used to develop the particle design, based on conservative assumptions (e.g., all fission gas generated during irradiation is released to the buffer) and conservative service conditions (e.g., simultaneous conditions of high temperature, high burnup, and high fast fluence). The fuel is manufactured and tested; both differential and integral tests are performed. The differential tests are used to develop and/or reduce uncertainties in the model parameters that are used to calculate fuel performance. The independent integral tests are used to validate the performance models. Finally, the validated design methods are used to predict the in-reactor fuel performance to confirm that the design meets the performance requirements.

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

As introduced in Section 2.3.2.3, several key failure mechanisms have been observed for SiC-based TRISO particles during irradiation and postirradiation heating tests; these mechanism are elaborated below.

3.2.3.1 Pressure-Vessel Failure

Pressure-vessel failure occurs when the tensile stress reached in the SiC (or ZrC) layer exceeds its strength. During irradiation, the inner and outer pyrocarbon layers undergo irradiation-induced shrinkage. This shrinkage puts the pyrocarbon layers into tension, but it also applies compressive forces to the SiC (or ZrC) layer, such that the layer should remain in compression (and not fail) if one or both pyrocarbon layers remains intact. However, if the shrinkage is too high and/or the irradiation-induced creep is too low, the resulting stresses may be sufficiently high to fail the pyrocarbon layers, and the SiC (or ZrC) layer could go into tension.

Pressure-vessel failure is of particular concern for high-burnup fuels, because of the possibility of CO formation during irradiation and the larger inventory of fission gases. For oxide kernels, oxygen is released when the heavy-metal atom undergoes fission. Some of the oxygen is consumed by oxidation of carbide phases and fission-products, but excess oxygen can react with the carbonaceous buffer to form CO under the thermochemical conditions that exist during irradiation. Oxygen potential in the particle can be controlled by use of a kernel with a low O/U ratio, by use of an oxygen getter such as ZrC, or by use of a kernel with a minor carbide phase.

3.2.3.2 Kernel Migration

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

3.2.3.3 Chemical Attack of the SiC Layer

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

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

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

3.3 Fission Product Transport Technology

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

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

3.3.1 Computer Codes

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

3.3.1.1 Core Performance Codes

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

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

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

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

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

3.3.1.2 Support Codes

SOLGASMIX-PV (Besman 1977): A thermochemical code that calculates equilibrium relationships in complex chemical systems by minimizing the free energy while preserving the masses of each element present for either constant pressure or volume. The code can calculate equilibria in systems containing a gaseous phase, condensed phase solutions, and condensed phases of invariant and variable stoichiometry. It has been used extensively to model kernel chemistry.

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

ABAQUS: A suite of industry-standard. general purpose, finite-element, structural analysis codes which can be used to perform full, deterministic, non-linear stress analysis. ABAQUS has been used to develop 2-D and 3-D pressure-vessel models for TRISO-coated fuel particles and to model various types of flaws, defects and structural abnormalities in the coating system.

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

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

3.3.2 Material Property Data

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

3.3.2.1 Radionuclide Release from Fuel Kernels

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

The present data base for fission gas release from heavy-metal contamination and from failed particles under irradiation is derived primarily from TRIGA measurements on fuel-compact matrix doped with uranium and on laser-failed fuel particles from capsules (Haire 1974, Homan 1978, Myers 1980, Myers 1984, Stansfield 1985, Myers 1987). Isothermal, inpile hydrolysis tests for the reaction of exposed kernels with water on LEU UCO fuel (HRB 17/18) were investigated at ORNL (Myers 1992). The temperature dependence of gas release from both unhydrolyzed and hydrolyzed LEU UCO fuel was addressed in the HFR B1 test which was conducted in HFR Petten in the Netherlands (Myers 1995, IAEA-TECDOC-978 1997). These tests indicate no strong burnup dependence for fission gas release from LEU UCO kernels up to a burnup of ~18% FIMA. Circumstantial evidence from the NPR-1/-2 irradiations of HEU UCO fuel to ~75% FIMA suggest a large burnup dependence (5-10x increases) at the higher burnup (Richards 1993). Limited German data for intermediate-to-high burnup UO_2 also suggest a burnup dependence for gas release (IAEA-TECDOC-978 1997).

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

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

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

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

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

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

3.3.2.2 Radionuclide Transport in Coatings

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

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

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

There are limited data regarding the diffusion of Pu, Am and Cm in PyC and SiC coatings on coated, uranium fuel particles (Baldwin 1978, Silva 1983, Minato 2000). The diffusion of these actinides in HTI and LTI PyC[13] appeared to be relatively rapid at high temperature. The actinides were quantitatively retained by SiC to at least 1400 °C; measurable releases were reported at 1600 °C, but the condition of the SiC coatings in these experiments is unknown (Minato 2000). There are no data available for Pu release from failed PuO_x particles or transport in PyC and SiC coatings on coated Pu particles.

Several researchers have investigated the potential of ZrC to retain metallic fission products although the data are very limited compared with that on SiC.

In Japan, the first generation of ZrC-coated fuel particles had a ZrC-C alloy layer, called "zirconium carballoy" with a C/Zr ratio of about 1.3. Retention of metallic fission products by the ZrC-C layer was determined to be poor (Fukuda 1979), which pointed out the importance of avoiding free carbon in the ZrC layer.

The most extensive and recent fission product retention data come from the series of post-irradiation heating tests performed by JAERI (Minato 1995, Minato 1997) on particles with stoichiometric ZrC coatings deposited using the JAERI bromide process. ZrC-coated UO_2 particles irradiated in capsule 80F-4A in the JMTR at about 900 °C to a burnup of 1.5% FIMA were subjected to heating tests at 1600 °C for 4500 hours, 1800 °C for 3000 hours, and 2000 °C for 100 hours to determine the release behavior of fission products. The fission products Cs-137, Cs-134, Ru-106, Ce-144, Eu-144, and Eu-155 were all released from the particles during heating. However, the fractional release of Cs-137 in the ZrC-coated particles was less than 1×10^{-3} even after heating at 1800 °C for 3000 hours, which is much better than the Cs-137 retention by SiC-coated particles. The effective diffusion coefficient of Cs-137 in the ZrC coating was estimated to be between 1×10^{-18} and 5×10^{-18} m^2/s at 1600 °C and between 2×10^{-18} and 1×10^{-17} at 1800 °C. The estimated Cs-137 diffusion coefficient for ZrC at 1600 °C is more than one order of magnitude lower than the Cs-137 diffusion coefficient determined for SiC. The estimated Cs-137 diffusion coefficient for ZrC at 1800 °C is more than two orders of magnitude lower than the corresponding Cs-137 diffusion coefficient for SiC (the SiC begins to degrade and become porous >1600 °C, hence the large increase in the apparent Cs diffusion coefficient).

However, the fractional release of Cs-137 in the ZrC-TRISO particles increased dramatically to more than 1×10^{-1} after heating at 2000 °C for 100 hours. The large increase in Cs-137 release

---

[13] Low Temperature Isotropic (LTI) PyC coatings are deposited from propylene or a mixture of propylene and acetylene at ~1300 °C. Earlier, High Temperature Isotropic (HTI) PyC coatings deposited from methane at ~2000 °C were also investigated, but they exhibited high failure rates at high fast fluences

from 1800 to 2000 °C was attributed to ZrC degradation due to reaction of the ZrC with CO (which had access to the ZrC coating layer as a result of failure of the IPyC coating layers between 1800 and 2000 °C) to form CO_2 and C (Minato 1998).

In contrast to the good Cs-137 retention by ZrC, Ru-106 release from the ZrC-TRISO UO_2 particles was very high with the fractional release being about 0.1 after heating for 4500 hours at 1600 °C and ~0.9 both after heating at 1800 °C for 3000 hours and heating at 2000 °C for 100 hours. These release fractions are approximately the same as for Cs-137 in heating tests of TRISO coated particles. However, no release of Ru-106 was reported in post-irradiation heating tests of TRISO particles heated at 1600 °C for as long as 500 hours and at 1800 °C for as long as 200 hours, which suggests that SiC is a better diffusion barrier to Ru-106 than is ZrC. However, it is important to note that the heating tests discussed above were performed in ZrC-TRISO UO_2 particles. In ZrC-TRISO UCO particles, it is expected that Ru-106 would be retained in the kernel so that Ru-106 release through the ZrC layer would not be an issue with this fuel particle design.

The fractional releases of Ce-144, Eu-154, and Eu-155 in the JAERI heating tests were also higher than for Cs-137, but the accuracy of the measured values was insufficient to permit a determination of the effective diffusion coefficient. Similarly, the inventory of Ag-110m in the particles heated in these tests was too small to permit any conclusion with respect to the retention of Ag-110m by ZrC.

The only data on diffusion of Ag-110m in ZrC reported in the literature is provided in (Chernikov 1986). Chernikov et al. performed heating tests at 1700 to 2200 °C on SiC-coated particles and ZrC-coated particles on which Ag-110m, Pm-147, Ba-133, and Ce-144 were deposited on the particle surfaces from a solution or were implanted on the particle surfaces by irradiating the test samples in contact with U-235. Stepwise removal and analysis of the coatings determined the distribution of the nuclides in the SiC and ZrC coating layers, and effective diffusion coefficients were estimated from the resultant data. Based on the estimated diffusion coefficients, Chernikov et al. concluded that ZrC is a more effective barrier than SiC to all of the investigated nuclides. However, the applicability of these results to the ZrC layer in irradiated ZrC-TRISO particles is unclear. Post-irradiation heating tests of ZrC-TRISO particles are needed to verify the effectiveness of ZrC, relative to SiC, as a diffusion barrier to Ag-110m.

UO_2*-C particles have exhibited outstanding fission product retention capability in post-irradiation heating tests performed by General Atomics (Bullock, 1984). In these tests, ten particles each of UO_2 TRISO, UC_2 TRISO, UCO TRISO, UO_2*-B, and UO_2*-C, which were irradiated in capsule HRB-15B to burnups in excess of 20% FIMA at about 900 °C, were heated for 10,000 hours at 1200, 1350, and 1500 °C. The UO_2*-C fuel particles were the only particle type to retain 100% of their fission product inventories, including highly mobile Ag-110m and Eu-154. The superiority of the UO_2*-C particles with respect to Eu-154 and Ag-110m retention was remarkable in that all 30 of the TRISO fuel particles released Eu-154 at levels from 15% and 100%, and 22 of 30 TRISO particles released Ag-110m at levels from 10% to 100%. Even though the number of particles in the test was small, the large difference in the results for UO_2*-C versus the conventional TRISO fuel particles would appear to clearly establish the superiority of the UO_2*-C design with respect to fission product retention.

The fission product retention by the UO_2*-B particles in the Bullock heating tests was not nearly as good as that of the UO_2*-C particles. These particles collectively released 27% of their Ag-110m inventory and 9% of their Eu-154 inventory at 1500 °C. However, the UO_2*-B particles were completely retentive of Cs-137 and Ce-144. Overall, the fission product retentiveness of the UO_2*-B particles at 1500 °C was superior to that of the UO_2 TRISO and UC_2 TRISO particle, and approximately equivalent to that of that of the UCO TRISO particles. Oddly, the UO_2*-B particles were the most highly releasing of Ag-110m in the 1200 and 1350 °C tests (2% and 19%, respectively).

3.3.2.3 Radionuclide Transport in Matrix and Graphite

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

The present correlations for fission metal diffusivities in core graphite are derived largely from laboratory measurements on unirradiated nuclear graphites and from profile measurements in various irradiated graphites (Hoinkis 1983, Myers 1984). The current correlation for Ag diffusivity in irradiated grade H-451 graphite was inferred from the measured Ag diffusivity in German A3 matrix (IAEA-TECDOC-978 1997).

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

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

The diffusivity of plutonium in unirradiated H-451 graphite has been measured up to 1350 °C (Godsey 1986), and the desorption pressure of Pu sorbed on H-451 has been measured up to 1350 °C (Fellows 1987). Based upon these measured transport properties, the release of Pu from the graphite into the primary coolant was predicted to be negligible for both normal operation and depressurized core conduction cooldown conditions. In addition, the sorption of PuC on

H-451 graphite (Tallent 1985) and uranium diffusion in H-451 graphite have been measured (Tallent 1985). The transport properties of actinides in compact-matrix material and of Am and Cm in graphite have not been measured.

3.3.3 Validation of Fission Product Transport Methods

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

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

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

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

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

The validity of the transient gas release model used to analyze core conduction cooldown transients has not been rigorously assessed. The validity of the reference design methods for predicting fission gas release from Pu cores has not been assessed.

The validity of the methods for predicting fission metal release from the core during normal operation have been assessed by applying the reference design methods to predict the observed metal release in Peach Bottom Core 2, FSV, irradiation capsules and in-pile loops (e.g., SSL1, SSL2, Idylle 03, four CPL2 loops, and R2 K13). Most of the available data are for the Cs isotopes with a small amount of Ag and Sr data. In general, the releases of fission metals were underpredicted by large factors, and in some cases, by more than a factor of 10. The cause of the underpredictions is ambiguous because the SiC defect fractions and the particle failure fractions are typically not well known. However, there is strong circumstantial evidence that the transport across the fuel compact/fuel element gap and the transport in the graphite web are not properly modeled.

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

The validations of the methods for predicting fission metal release during core conduction cooldown transients have not been assessed systematically. The validity of the reference design methods for predicting fission metal release and actinide release from Pu cores has not been assessed.

3.4 References for Section 3

ABAQUS, A General Purpose, Production Oriented Finite Element Code, Hibbitt, Karlsson, and Sorenson, Inc., 1989.

Alberstein, D., P. D. Smith and M. J. Haire, "Metallic Fission Product Release from the HTGR Core," Report GA-A13258, May 15, 1975.

Bairiot, H., L. Aerts, and E. Trauwaert, "Plutonium Coated Particle Development," Nucl. Tech., **23**, 240-255, 1974.

Baxter, A., D. McEachern, D. Hanson, and R. Vollman, "FSV Experience in Support of the GT-MHR Reactor Physics, Fuel Performance, and Graphite," Proceedings of the IAEA High Temperature Reactors and their Future Role," November 1994.

Bennett, R. G., and L. H. Menke, "User's Manual for NP-MHTGR FUEL Code," EGG-NPR-10249, INEEL, April 1992.

Bokros, J., and R. Price, Carbon, **3**, 503 (1966).

Besmann, T. M., "SOLGASMIX-PV, A Computer Program to Calculate Equilibrium Relationships in Complex Chemical Systems," ORNL/TM-5775, Oak Ridge National Laboratory, April 1977.

Bradley, J. S., and M. B. Richards, "Software Description and User's Manual for CAPPER Irradiation Capsule Fuel Performance Code," CEGA-002309, Rev. 0, CEGA, September 1992.

Bresnick, S., "MHTGR Fuel Process and Quality Control Description," DOE-HTGR-90257, Rev. 0, General Atomics, September 1991.

Bullock, R., "Toward Development of an injectable Thermosetting Matrix for HTGR Fuel Rods: I. Specimen Fabrication for Irradiation Testing," GA-A13851, General Atomic, April 1976.

Bullock, R., and S. Sterling, "Irradiation Performance of HTGR Fuel Rods with Diluted Thermosetting Matrices," Nucl. Eng. & Des., **44**, 109-123, 1977.

Bullock, R., "All-Ceramic Fuel Elements for the High-Temperature Gas-Cooled Reactor (A Review)," High Temperature - High Pressure, **12**, 569-633, 1980.

Bullock, R., "Full-Fluence Tests of Experimental Thermosetting Fuel Rods for the High-Temperature Gas-Cooled Reactor," Nucl. Tech., **52**, 246-259, 1981.

Bullock, R. E. and J. L. Kaae, "Performance of Coated UO_2 Particles with ZrC," J. Nucl. Mat., **115**, 69-83, 1983.

Bullock, R. E., "Fission-Product Release during Postirradiation Annealing of Several Types of Coated Fuel Particles," J. Nucl. Mat. **125**, 304-319, 1984,

Cadwallader, G. J., "SORS/NP1 Code Description and User's Manual," CEGA-002092, CEGA, September 1993.

Carbon Conference, "Extended Abstracts and Program, 16th Biennial Conference on Carbon," American Carbon Society, University of California, San Diego, July, 1983.

Carney, H., et al., "Alternate Fuel Cycle Evaluation Program - Fuel Cycle Characterization Study - HTGR Reprocessing Plant," GA-A15240, General Atomic, July 1979.

Chernikov, A. S., et al., "Fission Product Diffusion in Fuel Element Materials for HTGR," Proceedings of IAEA Specialists Meeting, Berkeley, 1985, IAEA IWGGCR 13, IAEA, 170-181, 1986.

Collins, J., M Lloyd, and R. Fellows, "The Basic Chemistry Involved in the Internal-Gelation Method of Precipitating Uranium as Determined by pH Measurements," Radiochimica Acta, **42**, 1987, p. 121.

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

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

Fukuda, K. et al., "Fission Product Diffusion in ZrC-Coated Fuel Particles," J. Nucl. Mat., **87**, 367-374, 1979.

GA-A14479, "HTGR Fuels and Core Development Program, Quarterly Progress Report for the Period Ending August 31, 1977," General Atomic, September 1977.

GA-A17700, "HTGR Spent Fuel Treatment Program - Economic Basis for HTGR Spent Fuel Treatment Decisions," GA Technologies, November 1984.

GA-LTR-23, "Safety Analysis Report: Use of UC_2 Fissile Fuel Particles in Fort St. Vrain Fuel Elements," General Atomic, September 1978.

Godsey, T. T., et al., "Diffusion of Plutonium in H-451 Graphite at 1000 to 1350°C," HTGR-85-033, ORNL, January 1986.

Goodin, D., "MHR-1 Compact Process Development Test Report," GA Document 910906, Rev. 0, General Atomics, November 1996.

Haire, M. J., and D. W. McEachern, "Gaseous Radioactivity Levels in the Primary Coolant of an HTGR," GA Report GA-A12946, General Atomic, October 1, 1974.

Hanson, D. L., "Plate-Out Phenomena in Direct-Cycle High Temperature Gas Reactors," EPRI Report 1003387, Electric Power Research Institute, June 2002.

Harmon D., and C. Scott, "Properties Influencing Particle Performance," Nucl. Tech. **35**, 343-352, September 1977.

Heit, W., et al., "Status of Qualification of High-Temperature Reactor Fuel Element Spheres," Nucl. Tech., **69**, 44-54, 1985.

Hoinkis, E., "Transport of Fission Products In Matrix and Graphite," Proceedings of a Colloquium held at the Hahn-Meitner-Institut, Berlin, from 9 to 11 November 1981, HMI-B372, June 1983.

Homan, F., H. Nabielek, and L. Yang, "Low-Enriched Fuel Particle Performance Review," USDOE Report GA-A14759, General Atomic, August 1978.

Homan, F. J., and M. J. Kania, "Irradiation Performance of HTGR Coated Particle Fuels With ZrC Coatings, ORNL/TM-9085, ORNL, January 1985.

Huschka, H., and P. Vygen, "Coated Fuel Particles: Requirements and Status of Fabrication Technology," Nucl. Tech., **35**, 238-245, September 1977.

IAEA-TECDOC-978, "Fuel performance and fission product behavior in gas cooled reactors," IAEA, November 1997.

IAEA-TECDOC-1198, "Current Status and future development of modular HTGR technology," IAEA, February 2001.

Journal of Nuclear Materials, "Proceedings of the 4th Inert Matrix Fuel Workshop," J. Nucl. Mat., **274**, January 1999.

Kasten, P. R., M. R. Corum, and P. L. Rittenhouse, "Research on Very High Temperature Gas Reactors," EPRI ER/NP-7372, EPRI, July 1991.

Ketterer, J. W., and B. F. Myers, "Capsule HRB-16 Postirradiation Examination Report, GA Document 908012, GA Technologies, September 1985.

Kirch, N., et al., "Storage and Final Disposal of Spent HTR Fuel in the Federal Republic of Germany," Nucl. Eng. & Des., **121**, 241-248, 1990.

Krautwasser, P., G. Begun, and P. Angelini, "Raman Spectra Characterization of Silicon Carbide Nuclear Fuel Coatings," J. Amer. Ceram. Soc., **66**, 424-434, 1983.irch

Hayashi, K., et al., "Some Topics on Improvement in Production Technology and Fission Product Behavior of HTTR Coated Particle Fuel, JAERI-Conf-96-010, 1996.

Lefevre, R. L. R., and M. S. T. Price, "Coated Nuclear Fuel Particles: The Coating Process and Its Model," Nucl. Tech.., **35**, 266-278, September 1977.

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

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

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

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

McEachern, D., R. Noren, and D. Hanson, "Manufacture and Irradiation of Fort St. Vrain Fuel," International HTR Fuel Seminar, Brussels, Belgium, February 2001.

McEachern, D., "Coated Particle Fuel for Transmutation," RDO-ROO-G-TRT-000121, Rev. A, General Atomics. September 2001.

Medwid, W., and A. Gillespie, "COMEDIE BD-1 Test Evaluation Report," DOE-HTGR-88552, Rev. 0, General Atomics, 1993.

Mehner, A. W., et al., "Performance of Ceramic HTR Fuel and Coating Materials under Normal and Transient Conditions," Proceedings of Gas-Cooled Reactors Today, Vol. 2, BNES, 139-144, 1982.

Miller, C., and W. Scheffel, "Postirradiation Examination and Evaluation of Peach Bottom FTE-13," GA Document 906939, General Atomics, November 1985.

Minato, K. et al., "Fission Product Release from ZrC-coated Particles during Postirradiation Heating at 1600 °C," J. Nucl. Mat., **224**, 85 – 92, 1995

Minato, K., et al., "Advanced Coatings for HTGR Fuel Particles against Corrosion of SiC Layer," J. Nucl. Mat., **246**, 215-222, 1997.

Minato, K. et al., "Fission Product Release from ZrC-coated Fuel Particles during Post-Irradiation heating at 1800 and 2000 °C," J. Nucl. Mat., **249**, 142-149, 1997.

Minato, K. et al., "Deterioration of ZrC-coated Fuel Particle Caused by Failure of Pyrolytic Carbon Layer," J. Nucl. Mat., 252, 13-21, 1998.

Minato, K., et al., “Fission Product Release Behavior of Individual Coated Fuel Particles for High-Temperature Gas-Cooled Reactors,” Nucl. Tech., **131**, 36-47, 2000.

Minato, K., et al., “Irradiation Experiment on ZrC-coated Fuel Particles for High Temperature Gas-Cooled Reactors,” Nucl. Tech., **160**, 2000.

Moormann, R., and K. Verfondern, “Methodik umfassender probabilistischer Shicherheitsanalysen fuer zukuenftige HTR-Anlagenkonzepte – Ein Statusbericht (Stand 1986), Band 3: Spaltproduktfreisetzung,” Juel-Spez-388, Vol. 3, KFA, 1987.

Morgan, M. T. and A. P. Malinauskas, “Cesium Release and Transport in BISO-Coated Fuel Particles,” Nucl. Tech., **35**, 457-464, September 1977.

Myers, B. F. and W. E. Bell, “Strontium Transport Data for HTGR Systems,” USAEC Report GA-A13168, General Atomic, December, 1974.

Myers, B. F., et al., “The Behavior of Fission Product Gases in HTGR Fuel Material,” USDOE Report GA-A13723, General Atomic, October 1977.

Myers, B. F., N. L. Baldwin, and W. E. Bell, “Fission Gas Release from Fuel Particles and Fuel Rods,” Nucl. Tech., **35**, 501-508, September, 1977.

Myers, B. F., and W. E. Bell, “Cesium Transport Data for HTGR Systems,” USDOE Report GA-A13990, General Atomic, September 1979.

Myers, B. F., and R. E. Morrissey, “Licensing Topical Report: The Measurement and Modeling of Time-Dependent Fission Product Release from Failed HTGR Fuel Particles under Accident Conditions,” USDOE Report GA-A15439, General Atomic, April 1980.

Myers, B. F., “Kernel Diffusion Coefficients,” GA Document 906031, Rev. 1, General Atomic, July 1981.

Myers, B. F., “Cesium Diffusion in Silicon Carbide during Post Irradiation Anneals,” HBK Report HBK-TN-01/84, FZJ, January 1984.

Myers, B. F., “Compilation of Data on Reactor Experiments for Use in Fission Product Transport Correlations,” HBK Report HBK-IB-05/84, September 1984.

Myers, B. F., “Fuel Design Data Manual,” GA Document 901866, Rev. F, General Atomics, August 1987.

Myers, B. F., “Effect of Water Vapor on the Release of Fission Gases from Uranium Oxycarbide in High-Temperature, Gas-Cooled Reactor Coated Fuel Particles,” J. Am. Ceram. Soc*.,* **75**, 686-693, 1992.

Myers, B. F., “The Effect of Water Vapor on the Release of Fission Gas from the Fuel Elements of High-Temperature, Gas-Cooled Reactors: A Preliminary Assessment of Experiments HRB-17, HFR-B1, HFR-K6 and KORA,” Report ORNL/TM-4294, Oak Ridge National Laboratory, 1995.

Nabielek, H., and B. F. Myers, "Fission Product Retention in HTR Fuel," Proceedings of Gas-Cooled Reactors Today, Vol. 2, BNES, 1982.

Nabielek, H., et al., "Development of Advanced HTR Fuel Elements," Nucl. Eng. & Des., 1990.

Naefe, P., and E. Zimmer, "Preparation of Uranium Kernels by an External Gelation Process," Nucl. Tech. **42**, 163, February 1979.

Nuclear Technology, "Special Issue on Coated Particle Fuels," **35 (2)**, September 1977.

Ogawa, T., et al., "Performance of ZrC-Coated Particle Fuel in Irradiation and Postirradiation Heating Tests," J. Am. Cer. Soc., **75**, 2985-2990, 1992,

Pelessone, D., "PISA, A Coupled Thermal-Stress Code for the Mechanical Analysis of Irradiated Fuel Particles – User's Manual," CEGA-002550, Rev. 0, CEGA, August 1993.

Petti, D., J. Maki, J. Buongiorno, R. Hobbins and G. Miller, "Key Differences in the Fabrication, Irradiation and Safety testing of U.S. and German TRISO-coated Particle Fuel and Their Implications on Fuel Performance," INEEL/EXT-02-00300, INEEL, April 2002.

Pfremmer, R. D., "Software Description and User's Manual for GT-MHR Fuel Performance Code, SURVEY," GA Document 911009, Rev. 0, General Atomics, April 2002.

Pfahls, O., "Verbesserung des Herstellverfahrens fuer HTR-Brennelemente. Abschlussbericht (Improved Manufacturing Process for HTR Fuel Elements. Final Report)," NUKEM-FuE-90013, NUKEM, Januar 1991 (*in German*).

Pollmann, E., J. Pelissier, C. Yust, and J. Kaae, "Transmission Electron Microscopy of Pyrocarbon Coatings," Nucl. Tech., **35**, 301-309, 1977.

Richards, M. B., "Status Report: Empirical Fuel Performance Model Based on PIE Data from Capsules HRB-21, NPR-1, NPR-2, and NPR-1A," CEGA-002952, Rev. 0, CEGA, October 1993.

Saito, S., et al., "Safety Requirements and Research and Development on HTTR Fuel," Report IWG-GCR/25 International Atomic Energy Agency, 1991.

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

Scott, C., D. Harmon, and J. Holzgraf, "Postirradiation Examination of Capsules P13R and P13S," GA-A13827, General Atomics, June 1976.

Silva, A. T., "Experimentalle Untersuchungen der Aktinidenfreisetzung aus Brennstoffteilchen fuer Hochtemperaturreaktoren," Juel-1833, KFA, March 1983.

Stansfield, O. M., W. A. Simon and A. M. Baxter, "Fuel Performance Models for High-Temperature Gas-Cooled Reactor Core Design," USDOE Report GA-A16982, General Atomic, September, 1983.

Stansfield, O. M., et al., “Advances in HTGR Fuel Performance Models,” USDOE Report GA-A17913, GA Technologies, February, 1985.

Sterling, S. A., “Cesium Release from Various Advanced HTGR Fertile Particle Designs,” GA Report GA-A15230, General Atomic, February 1979.

Steward, K. P., “Final Summary Report on the Peach Bottom End-of-Life Program,” USDOE Report GA-Al4404, General Atomic, 1978.

Stinton, D. P., et al., “Coating of Crystalline Nuclear Waste Forms to Improve Inertness,” J. Am. Ceram. Soc., **68**, 394-398, 1982.

Tallent, O. K., et al., “Vapor Pressure of Plutonium Carbide Adsorbed on Graphite,” ORNL/TM-9161, Oak Ridge National Laboratory, September 1984.

Tallent, O. K., et al., “Uranium Diffusion in H-451 Graphite,” Nucl. Tech., **68**, 336-343, 1985.

Tzung, F., “TRAFIC-FD, A Finite Difference Program to Compute Release of Metallic Fission Products from an HTGR Core: Code Theory and Users Manual,” CEGA-001904, CEGA, September 1992.

Tzung, F., COPAR-FD, A Finite Difference Program to Compute Release of Metallic Fission Products from Coated Particles: Code Theory and Users Manual,” CEGA-002098, Rev. N/C, CEGA, September, 1992.

Voice, E. H., and D. N. Lamb, “The Deposition and Structure of Pyrolytic Silicon Carbide,” DP-667, Dragon Project, 1969.

4. Provisional Fuel Requirements

4.1 *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.

4.1.1 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. 4-1, 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 4-1. The GT-MHR fuel element and its components are shown in Fig. 4-2. The following subsections provide brief descriptions of the coated fuel particles, fuel compacts, and fuel-element graphite blocks.

4.1.1.1 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 particle loadings are varied with location in the core, in order to optimize reactivity control, minimize power peaking, and maximize fuel burnup. The buffer, inner pyrolytic carbon, silicon carbide, and outer pyrolytic carbon layers are referred to collectively as a TRISO 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 TRISO particle design parameters for the GT-MHR (and, by inference, for the VHTR) are given in Table 4-2; advanced coated-particle designs are described in Section 4.3.

4.1.1.2 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 serve as an additional barrier to the release of metallic fission products. Compact design parameters are given in Table 4-3.

4.1.1.3 Fuel Element

Each fuel element is made from a machined graphite 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 (31.22 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 the 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.

4.1.2 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.

4.1.3 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, which were adopted here for the purpose of establishing goals for advanced fuel development, are compared with the conditions for the 850 ^{o}C GT-MHR in Table 3-4.

The peak fuel temperature in the commercial GT-MHR with an 850 ^{o}C core outlet temperature is expected to be ~1250 ^{o}C for normal operation and <1600 ^{o}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 ^{o}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 ^{o}C increase in core outlet temperature translates into about a 50 ^{o}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 ^{o}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 will 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.

4.2 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. To reiterate, 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. 4-3 (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 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 licensing of 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 development 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 a 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 overall 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 4-5, and the provisional metal release limits are shown in Table 4-6. 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 VHTR 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.

4.3 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. 2-3 for a conventional TRISO particle, the coating layers have specialized purposes (Section 2.3.2) but, in composite, provide a high-integrity pressure vessel which is extremely retentive of fission products (e.g., Bullock, 1994). 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 to design burnup, analytical design methods must be used to predict in-core fuel performance and to demonstrate compliance with fuel performance and fission product release criteria.

As introduced in Section 2.3.2, the current design methods used to predict fuel performance in prismatic-core HTGRs (e.g., IAEA-TECDOC-978 1997) consider eight potential failure mechanisms[14] which are illustrated schematically in Fig. 2-5:

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. Failure of the SiC coating due to kernel migration in the presence of a thermal gradient.
6. Failure of the SiC coating caused by fission product/SiC interaction.
7. Failure of the SiC coating by thermal decomposition.
8. Heavy-metal dispersion during SiC coating deposition and subsequent accelerated SiC corrosion during irradiation.

The conventional, TRISO-coated, fissile and fertile particle designs specified for the GT-MHR (Munoz 1994) and summarized in Table 4-2 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 >1000 °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 4-7 and 4-8, 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

---

[14] Irradiation-induced failure of the IPyC is not explicitly modeled in the current core survey codes; the IPyC is conservatively neglected. It is anticipated that this conservatism will be removed from the core survey codes in the near future. Irradiation-induced failure of the IPyC is explicitly modeled in the particle design codes, such as PISA (Section 3.3.1.2).

need to be revised and embellished a number of times, certainly after the screening and qualification test phases are completed.

4.4 References for Section 4

Bullock, R., and C. Young, “TRISO Fuel Particle Design Basis,” DOE-GT-MHR-100225, General Atomics, 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., et al., “Stoichiometric Effects on Performance of High-Temperature Gas-Cooled Reactor Fuels from the U-C-O System,” Nucl. Tech., **35**, 428-441, September, 1977.

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,” GA Document 906939, Rev. N/C, General Atomics, November 1985.

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

Shenoy, A., “Gas Turbine-Modular Helium Reactor (GT-MHR) Conceptual Design Description Report,” GA Document 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, 1995.

Table 4-1. 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 4-2. GT-MHR 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 4-3. GT-MHR Fuel Compact Design Parameters

Parameter	Design Value
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 4-4. Provisional Service Conditions for VHTR Fuel

Performance Parameters	**GT-MHR**	**VHTR**[15]
Core outlet temperature	850	1000
Core power density	6.6	6.6
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
- accident conditions	<1600	<2000

---

[15] Bounding values for the purpose of planning this advanced fuel technology program; a design goal for the VHTR is optimize the core and plant design such that GT-MHR peak temperature limits can also be met (or nearly so) with a 1000 ^oC core outlet temperature.

Table 4-5. 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)	$[\leq 6.0 \times 10^{-4}]$	$[\leq 3.0 \times 10^{-4}]$	$[\leq 1.2 \times 10^{-3}]$

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

Table 4-6. Provisional Fission Metal Release Limits

Reactor Plant	**Type**	**COT[16] (°C)**	**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	$[1.0 \times 10^{-5}]$	$[1.0 \times 10^{-4}]$	$[2.0 \times 10^{-4}]$	$[2.0 \times 10^{-3}]$

---

[16] COT = core outlet temperature

Table 4-7. Fuel Specification for UO_2* Particles

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

Fertile Kernel		
Kernel Feature	Specification	
Composition	UO_2	
Diameter	500 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]
Fertile TRISO Coating		
Buffer Layer	65	<1.0
IPyC	35	1.85 – 1.9
SiC	35	3.2
OPyC	40	1.85 – 1.9

Table 4-8. Fuel Specification for TRIZO Particles

Fissile Kernel	
Kernel Feature	Specification
Composition	UCO
Diameter	350 μm
Density	>10
O/U Ratio	[<0.15]

Fissile 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

Fertile Kernel	
Kernel Feature	Specification
Composition	UCO
Diameter	500 μm
Density	>10
O/U Ratio	[<0.15]

Fertile TRISO Coating		
Coating Description	Coating Thickness	Coating Density (g/cc)
Buffer Layer	65	<1.0
IPyC	35	1.85 – 1.9
ZrC	35	[6.7]
OPyC	40	1.85 – 1.9

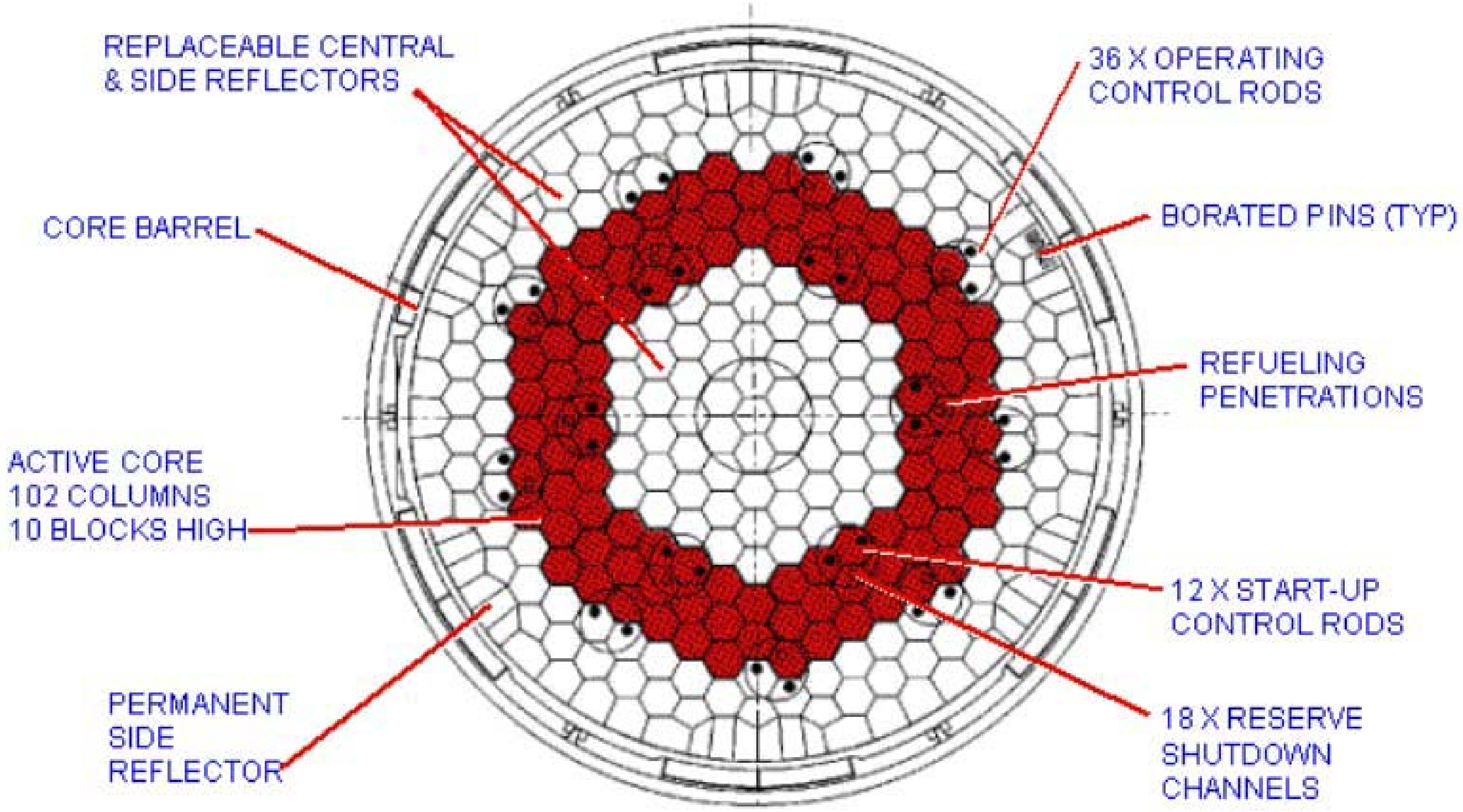


Figure 4-1. GT-MHR Core Layout

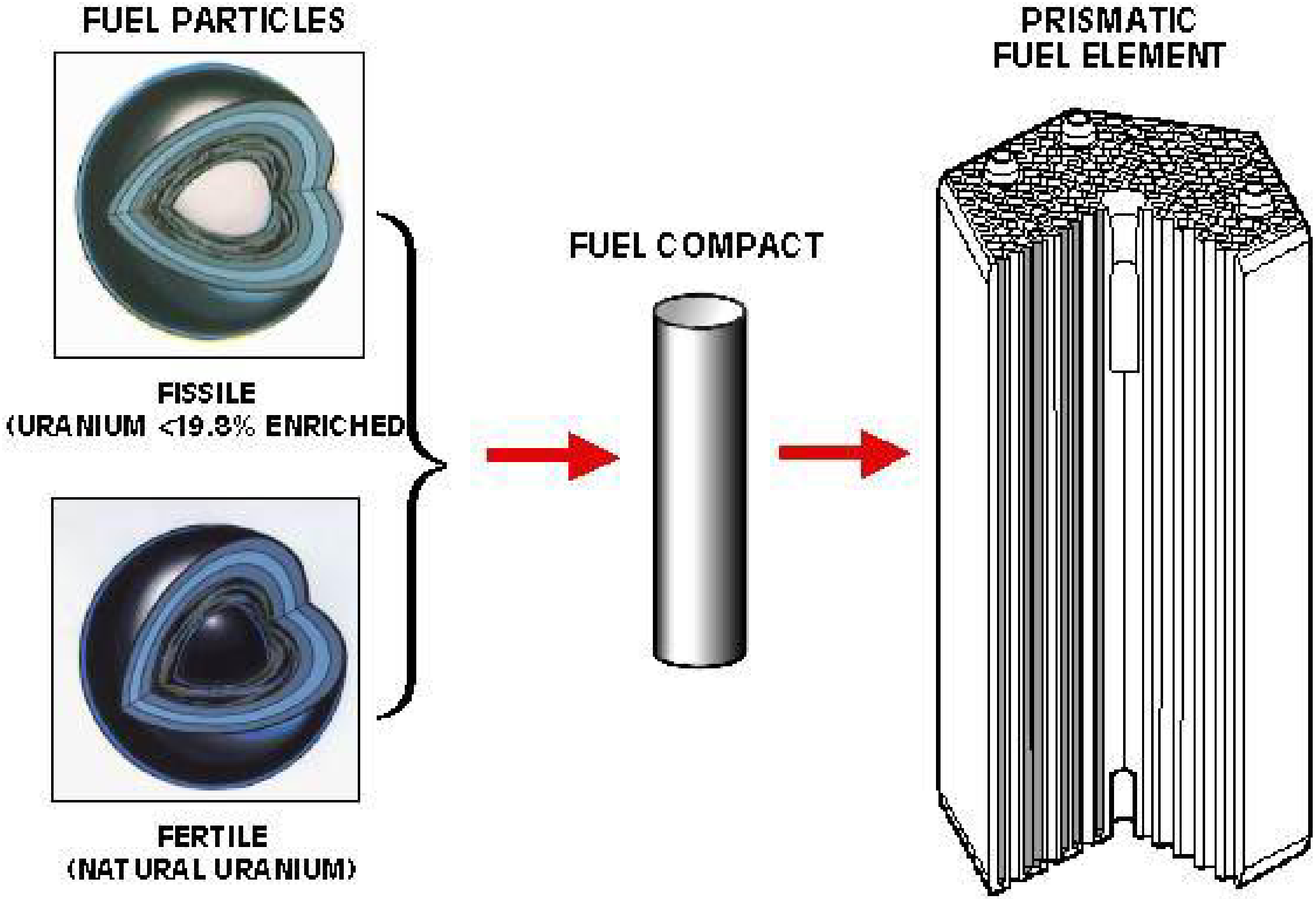


Figure 4-2. Fuel Element Components

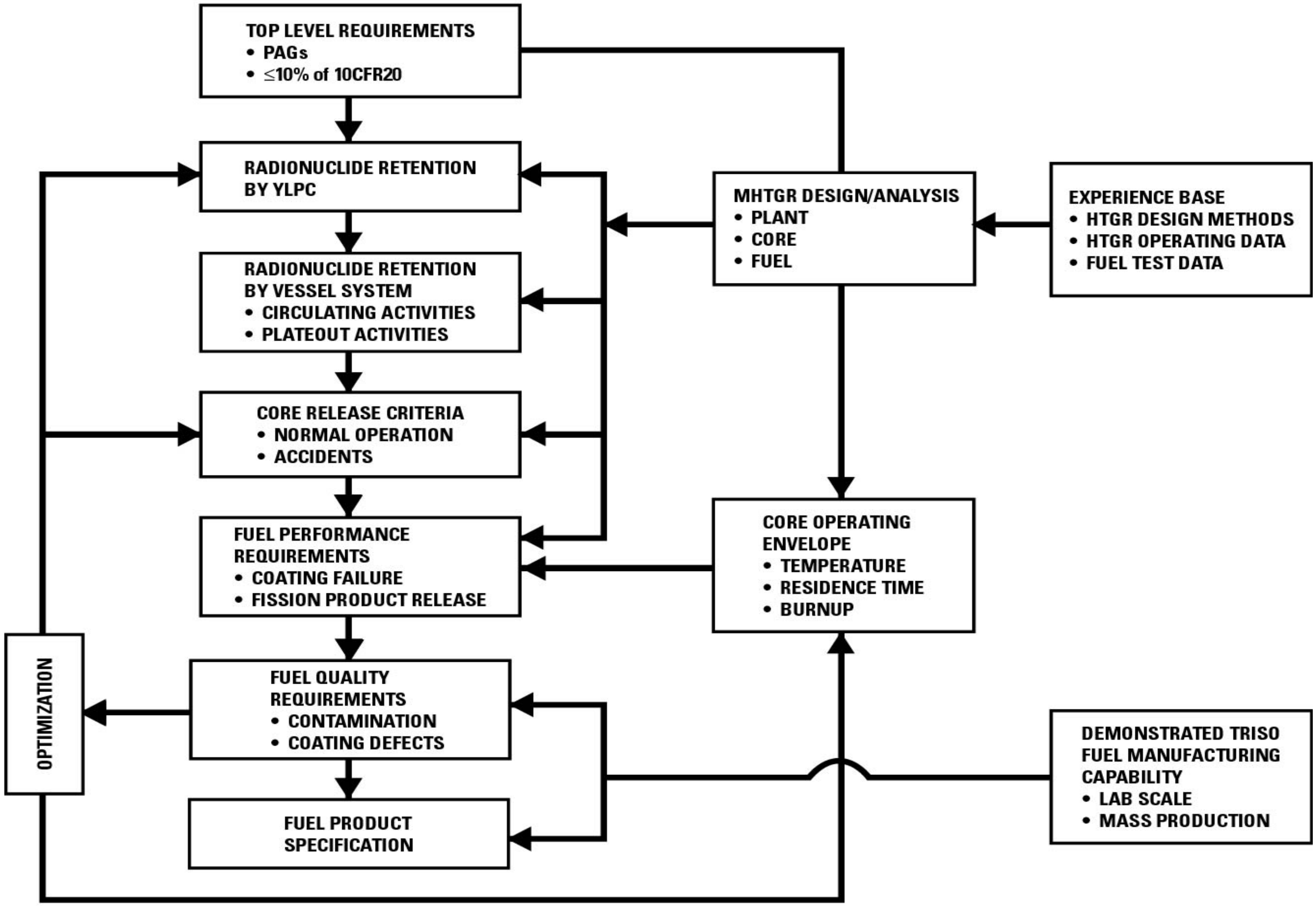


Figure 4-3. Logic for Deriving Fuel Quality Requirements

5. Design Data Needs

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

5.1 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. 5-1.

5.2 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 5-1, 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.

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 5-1 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 6.

The fuel DDNs assigned to this plan largely address the fabrication and testing of advanced coated particles upon the assumption that the AGR Program will complete the development and qualification of conventional TRISO UCO particles. Likewise, the fission product DDNs assigned to this plan largely address transport in kernels, particle coatings, and fuel-compact matrix. As indicated in Table 5-1, 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. Certain DDNs (e.g., tritium transport in core materials, DDN VHTR.03.10) are not currently addressed in the AGR Plan; hence, they are addressed herein. The DDNs related to tritium production and transport are of particular interest to a VHTR for hydrogen production because the tritium produced in the primary coolant circuit can permeate through the heat exchangers and contaminate the hydrogen product (Gainey 1976).

5.3 References for Section 5

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

[DDN Procedure], “DOE Projects Division Program Directive #16: HTGR PROGRAMS - Design Data Needs (DDNs) Interim Procedure,” PD#16, Rev. 1, Plant Design Control Office, February 1986.

Gainey, B. W., “A Review of Tritium Behavior in HTGR Systems,” GA-A13461, General Atomic, April 1976.

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,” IAEA, November 1997.

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

Table 5-1. 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	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	**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
VHTR.03.16	Fission Product Transport in a Vented Low-Pressure Containment	AGR, RF
VHTR.03.17	Decontamination Efficiency of Depressurization Train Filter	AGR

DDN No.	DDN Title	Data Source
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

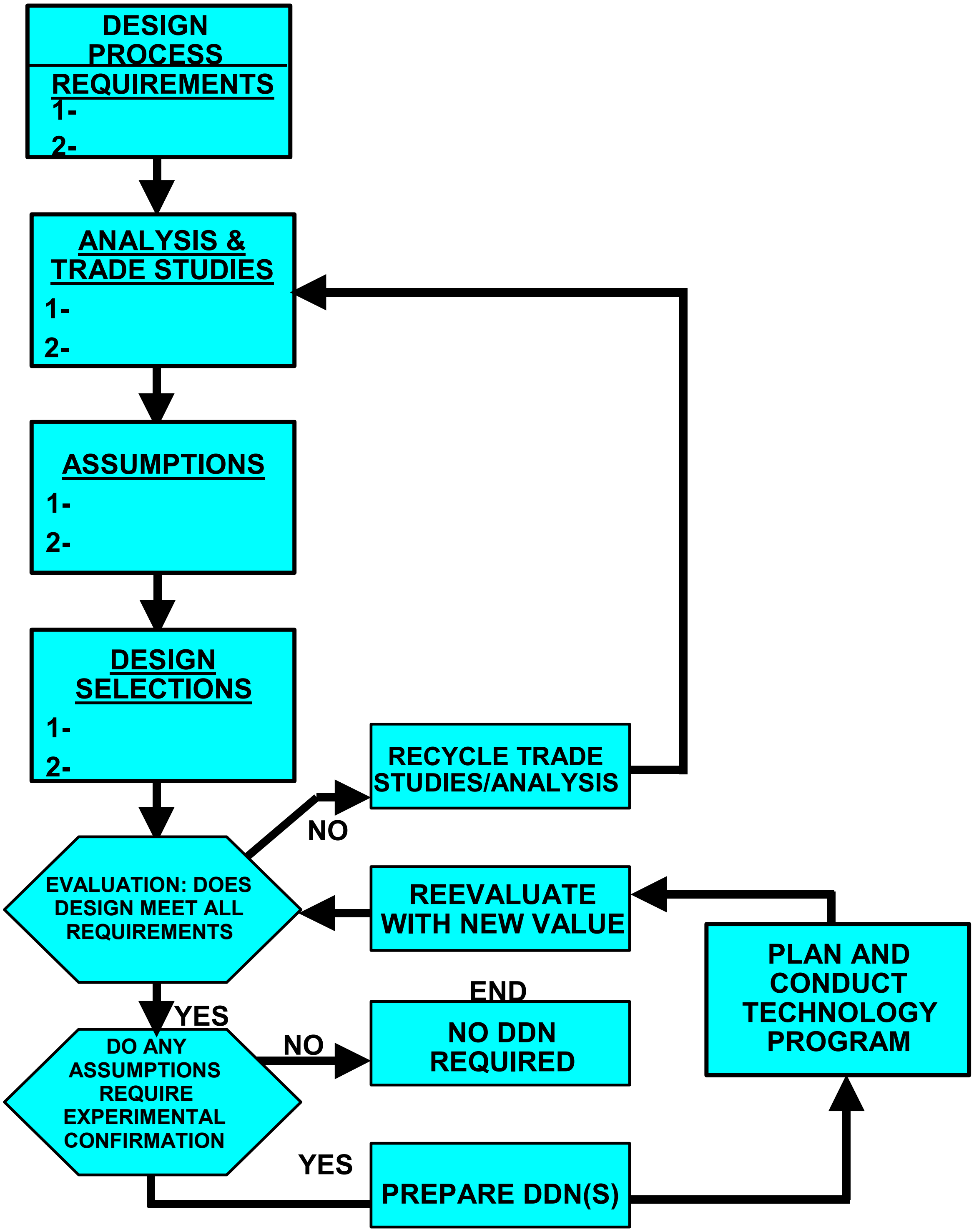


Figure 5-1. Process for Identifying DDNs

6. Technology Development Programs

The formal goal of this development program is to contribute to the resolution of the fuel/fission product DDNs (Section 5) 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 meets performance requirements in a core characterized by sustained temperatures up to 1400 °C during normal operation and temperatures up to 2000 °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 °C and the postirradiation heating behavior up to 2000 °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 postirradiation heating of compacts from these capsules at 1600 and 2000 °C.
4. Successful scale-up of the fabrication process to produce the reference advanced fuel particle design will be demonstrated by irradiation capsules VHTR-3 and VHTR-4, and subsequent postirradiation heating of compacts from this capsule at 1600 and 2000 °C. Should the performance of the reference advanced fuel particles be unacceptable in these tests, further process development and irradiation testing that is beyond the scope of this program plan will likely be required.
5. Screening of advanced (“exotic”) particle designs will be based upon irradiation capsule VHTR-6 and the subsequent postirradiation heating of compacts from this capsule at 1800, 2000, and 2200 °C.
6. Most postirradiation 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.

If accidents in addition to those anticipated here are determined to be included in the licensing basis, then additional tests with reference fuel may be required to quantify the kinetics of particle failure and fission product release as a function of time, temperature and oxidant concentration.

This umbrella development plan was preceded by an earlier screening plan (Hanson 2003, PC-000510/0) which focused on the irradiation and accident simulation tests needed to select and qualify an advanced fuel for the VHTR. The workscope in the screening plan is included and embellished in this plan.

A Work Breakdown Structure is presented in Table 6-1 to organize and manage these advanced fuel development activities. The Fuel Design tasks **(WBS 1.0)** define the design requirements and programmatic requirements that 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.

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 include measurements on test fuel performed during irradiation, post-irradiation examination, and accident condition testing. The Radionuclide Transport tasks **(WBS 2.3)** cover 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.

6.1 Fuel Design

The Fuel Design tasks **(WBS 1.0)** define the design requirements and programmatic requirements that provide the top-level goals and context for the Fuel Development tasks described below. Formally, these tasks are design activities rather than technology development activities. They are included here because there is currently no program to design a VHTR Demonstration Module with a reactor core utilizing advanced coated-particle fuel. Nevertheless, these design activities are essential to define the quantitative fuel performance requirements and the VHTR-specific Design Data Needs that drive the advanced fuel technology development program described in the following subsections. In the event that such an advanced VHTR core design program were to evolve from the current NGNP preconceptual design activities, these tasks would likely be transferred to that program.

6.1.1 Design Data Needs

WBS 1.1: A Design Data Need is a summary statement of a technology development need in a prescribed format (DDN Procedure 1986). In addition to a summary of the data needed, the standard DDN format includes the following programmatic information: (1) designer's alternatives, (2) selected design approach and explanation, (3) schedule requirements, (4) priority and (5) fallback position and consequences of nonexecution. (The contents of these subsections are intuitive from the headings.)

The DDNs related to fuel performance and fission product transport for a VHTR are listed in Table 5-1. Due to funding constraints, VHTR-specific DDNs were not developed in detail prior to preparing the initial issue of this advanced fuel development plan. Rather, the data needed were inferred from existing fuel/fission product DDNs, principally those for the direct-cycle GT-MHR (DOE-GT-MHR-100217 1996). Under this task, a complete set of VHTR-specific, fuel/fission product DDNs will be prepared. An initial issue will be prepared at the start of this development program (assumed here to be 7/1/04), and the DDNs will be updated at the start of the preliminary and final design phases; these latter two issues of the DDNs will be progressively more design specific and quantitative as the VHTR design matures as a result of the NGNP program.

6.1.2 Fuel Development Plan

WBS 1.2: This advanced fuel technology development plan will be updated periodically. The first update will be initiated at the start of the program, and it will reflect the comprehensive set of VHTR-specific, fuel/fission product DDNs (Section 6.1.1) and external review comments on this initial issue. It will also include a more detailed integration of this advanced fuel program with the baseline AGR Program. This plan will be updated early in the preliminary and final design phases; these latter two issues of the plan will be based upon the updated DDNs prepared at the start of preliminary and final design phases, respectively. The updated plans will also reflect the then available results from the ongoing international coated-particle fuel development programs, especially the baseline AGR Program.

6.1.3 Fuel Specifications

WBS 1.3: Fuel product specifications and fuel process specifications will be prepared to provide the technical basis for the fuel process development tasks described in Section 6.2. The fuel product specification (e.g., Munoz 1994) defines and controls the required as-manufactured fuel attributes as well as the QC methods used to demonstrate compliance, and the fuel process specifications (e.g., DeVelasco 1987) defines the process conditions to be used for the fabrication of fuel kernels, coated particles and compacts. The process specifications will be augmented by separate equipment and raw materials specifications as required. The existing specifications for the fabrication of conventional TRISO particles will be used as appropriate or modified as required; at a minimum, they will serve as templates for the preparation of new specifications for advanced fuels. A technical support document giving the bases for the product and process specifications will also be prepared and updated periodically.

Analytical studies will be performed of candidate advanced particle designs with the emphasis on UO_2^* and TRIZO UCO. Structural analyses with one or more of the particle design codes described in Section 3.X will be performed to provide the technical basis for specifying the mechanical designs of these advanced particles (e.g., kernel diameter, buffer thickness, etc.). Sensitivity studies will be performed to help establish tolerance limits on key variables. Thermochemical analyses will also be performed to determine the kernel composition for TRIZO UCO and to determine the Zr content in UO_2^*. The results will be documented in a particle design report (e.g., Richards 2002).

As with the DDNs and development plan, an initial issue of the fuel product specification and fuel process specification will be prepared at the start of this development program. The initial product specification will be largely predicated on the specification for the GT-MHR (Munoz 1994), and the initial process specification will be based largely upon the published process data for UO_2^* and ZrC coatings summarized in Section 3. These specifications will be updated at the start of the preliminary and final design phases; the latter two issues of the product specification will be progressively more design specific and quantitative as the VHTR design matures, and the latter two issues of the process specification will reflect the process development studies performed under this program (Section 6.2).

6.1.4 Model Development

WBS 1.4: As discussed in Section 3, analytical methods have been developed to predict TRISO fuel performance and fission product transport under irradiation and under postulated accident conditions. Such models are necessary to predict fuel performance and fission product transport in the reactor as well as in irradiation tests and in postirradiation heating tests. Initially, the existing models for conventional TRISO particles will be used for advanced particle designs with particle-specific material property data (e.g., ZrC strength data) as available. (Based upon the data review summarized in Section 3, there are few published data for the advanced particle designs of interest here.) As test data are generated under this program, fuel performance models and fission product transport correlations will be derived from the data. Models for UO_2^* and TRIZO UCO will be derived at the start of preliminary design and again at the start of final design phases. The models developed in this subtask will be documented and controlled in future issues of the Fuel Design Data Manual (Myers 1987). A technical support document giving the bases for the FDDM will also be prepared and updated periodically.

6.1.5 Design Methods Validation

WBS 1.5: Again as discussed in Section 3, the analytical methods developed to predict fuel performance and fission product transport under irradiation and under postulated accident conditions must be formally validated to demonstrate that they have the required predictive accuracies. The first subtask in this area will be to prepare a methods validation plan (e.g., Maneke 1988) early in preliminary design which will describe how generic test data from past programs and on-going technology programs in combination with particle-specific data from this program will be used to validate the design methods. The final multi-year subtask, which will be completed in FY2015, will utilize the available test data, especially data from VHTR-7, to assess the validity of the design methods, including particle-specific component models (Section 6.1.4).

6.2 Fuel Process Development

As stated previously, this advanced fuel program is an incremental program with the AGR fuel program providing the base technology; in the present context, it is assumed that the AGR Program will optimize the existing processes for fabricating LEU UCO kernels, for applying TRISO coatings to oxidic kernels, and for fabricating thermosetting resin-based fuel compacts. Moreover, the process development 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. The workscope and the associated cost and schedules estimates given herein do include the replication of the large-diameter coater that will designed and constructed by the AGR Program, and they also include the design and the construction of Zr halide feed system that can be scaled up for use with a production coater.

6.2.1 Kernel Fabrication

WBS 2.1.1.1: The capability to make relatively large (i.e., kilogram) quantities of UCO kernels by the reference internal gelation process currently exists at BWX Technologies (BWXT). Only a few improvements to the UCO fabrication process at BWXT are needed:

- Upgrade the method of dispersing carbon in the acid-deficient uranyl nitrate and establish the process parameters to ensure complete wetting of the carbon

- Establish the reference set of process parameters for reliable fabrication of 350 μm kernels
- Develop an environmentally acceptable substitute for trichloroethylene as a kernel forming media

These improvements will be accomplished within the scope of the AGR Fuel Development and Qualification Program. Consequently, the only workscope currently identified for this work area is liaison with the AGR Program. In the event that unanticipated complications were to arise when applying a TRIZO coating system to a UCO kernel, additional workscope would be added.

WBS 2.1.1.2: The UO_2 kernel fabrication process is a mature process that has been used worldwide to fabricate large quantities of kernels. Consequently, no significant development of this process is necessary. The ZrC coating process development described in Section 6.2.2.1 is applicable to deposition of a thin ZrC coating on pyrocarbon-sealed UO_2 kernels (to make UO_2*-C kernels) as well as to deposition of a ZrC layer in lieu of a SiC layer in the TRISO coating system. A modest addition to the ZrC coating development effort described in Section 6.2.2.1 will be needed to finalize the coating conditions for co-depositing ZrC in the buffer layer in the UO_2*-B particles. The progress of international organizations (NFI, INET, and PBMR) who are actively engaged in the production of UO_2 kernels will also be monitored. In the event, that unanticipated complications were to arise when applying PyC seal coats or ZrC overcoats to standard UO_2 kernels, additional workscope would be added.

WBS 2.1.1.3: As discussed in Section 3, other kernel compositions and kernel additives (e.g., getters) have been investigated and some work on nonconventional kernels (e.g., nitride kernels for fast reactor fuel) is on-going. Whether this program should aggressively explore exotic kernel compositions is uncertain at this writing. The argument could be made that the UCO kernel has been demonstrated to be an effective oxygen getter for suppression of CO to high burnups such that no further development of oxygen getters is necessary. On the other hand, past attempts to use kernel additives to chemically bind volatile radionuclides in the kernel (e.g., the addition of $xAl_2O_3.ySiO_2$ to getter Cs isotopes) have not been successful. Consequently, the only workscope currently identified for this work area is monitoring the progress of international organizations (ORNL, CEA, etc.) who are actively investigating nonconventional kernel compositions (e.g., nitrides for fast reactor fuels).

WBS 2.1.1.4: Sufficient quantities of UCO and UO_2 kernels will be manufactured to support the planned coating process development activities and to provide the feedstock for the fabrication of test specimens for the planned irradiation program (WBS 2.2). Given the nature of the kernel fabrication process, it is more efficient to manufacture a sufficient quantity of kernels in a single production campaign than to manufacture small quantities in multiple campaigns spread over a number of years. Consequently, this program will follow the lead of the AGR program and manufacture sufficient UCO and UO_2 kernels in a single campaign early in the program to support the entire planned development program. Like the AGR Program, the existing kernel line at BWXT, based upon the internal gelation process, will be used when available to this program. The same equipment can be used to fabricate both UCO and UO_2 kernels because the various steps in the process are essentially the same: in the fabrication of UO_2 kernels, the partial carbothermic reduction of UO_2 kernels to produce UCO kernels is simply eliminated.

6.2.2 Coated Particle Fabrication

The coating process development defined in this fuel development plan is limited to the process development needed to (1) deposit a ZrC coating (in lieu of a SiC coating) in a standard TRISO particle and to deposit the ZrC in UO_2* particles, and (2) produce more exotic fuel particle design candidates for irradiation in screening capsule VHTR-6.

WBS 2.1.2.1: The AGR Program will optimize and qualify the coating processes for deposition of conventional TRISO coating layers. Consequently, the only workscope currently identified for this work area is liaison with the AGR Program. In the event, that unanticipated complications were to arise when applying a TRISO coating system to a UO_2^* kernel, additional workscope would be added

6.2.2.1 UO_2^* Coating Development

The convention adopted here is to consider the application of the PyC seal coat and ZrC overcoat as part of the UO_2^*-C kernel fabrication process (**WBS 2.1.1.2,** Section 6.2.1). For the UO_2^*-B particle, process development will be conducted to determine the optimal process and coating conditions for co-depositing low-density pyrocarbon and ZrC. The greatest challenge will likely be to identify the best process for introducing a suitable volatile Zr compound into the coater and to develop appropriate equipment and procedures for large-scale production of these coatings (see the next subsection).

6.2.2.2 ZrC Coating Development

WBS 2.1.2.2: Based on past ZrC process development experience (see Section 3.1.2), the JAERI bromide process and the LANL $ZrCl_4$ sublimation process have been the most reliable with respect to controlling Zr-halide supply to the coater, but these processes have been demonstrated only in laboratory-scale coaters. An engineering evaluation will be performed to assess the viability of these processes for reliable supply of Zr-halide to a production-size coater. It is assumed that a reference process selection will be made between the bromide process and the $ZrCl_4$ sublimation process, but it is possible that neither process will be determined to be viable for scale-up, in which case an alternate approach will need to be developed.

The initial ZrC coating development effort will be comprised of parametric studies in a laboratory-scale coater to optimize the reference coating process and to prepare samples for irradiation testing. The process parameters to be investigated in these parametric studies will include coating temperature, CH_4 concentration, H_2 concentration, and Zr-halide concentration. The parametric studies will be limited to a relatively narrow range that has been shown by previous experimental work (see Section 3.1.2) to produce ZrC-TRISO fuel particles with good irradiation performance and fission product retention. The results of these parametric studies will confirm and supplement the results of the previous work performed by JAERI and LANL, and the supplemented database will serve as a point of departure for scale up of the process to a production-size coater.

The parametric studies will produce a reference ZrC-TRISO particle and variants having a range of ZrC properties for irradiation testing and for post-irradiation heating tests. These tests will yield data that can be used to correlate process parameters with the mechanical and chemical stability, including oxidation characteristics, of the ZrC coatings and with their fission product retentiveness. The tests will also yield first-time data concerning the irradiation and accident condition performance of ZrC-TRISO UCO fuel particles. It is anticipated that ZrC-TRISO particles having ZrC layers with a range of C/Zr ratios, densities, and morphologies will be irradiated in screening capsule VHTR-2 and potentially in "piggy-back" samples in capsule AGR-1, which is to be irradiated under the AGR Program.

The ZrC coating on the UO_2*-C kernel will be fabricated in a laboratory-scale coater using the reference ZrC coating process and the reference set of ZrC coating conditions developed from the ZrC-TRISO parametric coating studies. The TRISO-coatings on the UO_2* kernels will be deposited using the reference set of coating conditions established by the AGR program, except for the buffer coating layer in the UO_2*-B particle. A limited number of coating runs will be needed to finalize the process parameters to be used to co-deposit the ZrC in the buffer of the UO_2*-B particles. The UO_2*-C and UO_2*-B particles will be irradiated in screening capsule VHTR-1 and potentially in "piggy-back" samples in capsule AGR-1.

The next phase of the ZrC coating development effort will be to scale up the reference ZrC coating process. The initial task in this phase will be to design and construct a Zr-halide feed system for the reference coating process. It is assumed that the resultant Zr-halide feed system will be interfaced with a large coater of the same basic design as that selected for use in the AGR Program. This coater may be the same coater used in the AGR Program, or it may be a replicate (with modifications, as appropriate). For cost estimating purposes, it is assumed that the production coater will be a replicate of the AGR Program coater.

It is expected that a reference set of coating conditions for depositing the standard TRISO coating layers in the large coater will have been established by the AGR Program, so further coating process development will be needed in the large coater only to scale-up the reference ZrC coating conditions established in the laboratory-scale coater. Success of the scale-up will be determined by verifying the uniformity of the product made in the large coater and by comparing the density, morphology, and microstructure of the ZrC coatings deposited in the large coater with the reference ZrC coatings fabricated in the small coater and tested in capsules VHTR-1 and VHTR-2. This scale-up process will be the same regardless of whether the ZrC-TRISO particle or a UO_2* particle is selected as the reference VHTR fuel particle based on their respective performance in VHTR-1 and VHTR-2 and in post-irradiation heating tests.

Reference fuel particles and a few variants (representing a change in one or more key process parameters) will then be fabricated in the large coater for irradiation testing in capsules VHTR-3 and VHTR-4 and for post-irradiation heating tests. If the performance of the reference fuel is verified in these tests, additional reference fuel particles will be fabricated for irradiation in capsule VHTR-5. Conversely, the data from the VHTR-3 and VHTR-4 tests may indicate the need for modifications in the reference ZrC coating process. The cost and schedule for the plan assumes that some minor optimization of the process may be required. However, should the results of these tests indicate that substantial changes to the fabrication process are required, further coating development and irradiation testing beyond the scope of this plan will be required (i.e., reprogramming will be required).

6.2.2.3 Development of Alternate Fuel Designs

WBS 2.1.2.3: In the event that neither UO_2* nor ZrC-TRISO particles exhibit acceptable irradiation performance in screening capsules VHTR-1 and VHTR-2, it may be necessary to go to an alternate fuel design. Realistically, these candidates would likely be limited to TRISO-like designs that include different types of oxygen and/or fission product gettering layers. Such designs might include particles with different combinations of SiC, ZrC, and SiC+C or ZrC+C layers. Development of a fuel design composed of some combination of SiC, ZrC, and SiC+C ZrC+C would not require significant additional coating process development because the coating processes to deposit all of these coating types will already have been established (following completion of the ZrC development described in Section 6.2.2.1).

However, other more exotic advanced fuels have been, or are being investigated for use in various reactor designs. These include, for example, new fuels for fast gas-reactors in which pellets of fuel are coated with a material such as titanium nitride as an alternative to graphite.[17] Also, CVD NbC-coated uranium oxide fuel and binary carbide fuels of (U,Zr)C were investigated in the KIWI and NERVA nuclear rocket propulsion programs in the 1960's[18]. Uranium bearing, solid-solution tri-carbide fuels such as (U, Zr, X)C, where X = Nb, Ta, Hf, or W are currently under development at the Innovative Nuclear Space Power and Propulsion Institute at the University of Florida for advanced space power and propulsion applications[19]. According to this reference, the presence of non-uranium carbides in the tri-carbide fuel allows for gradient coating of fuel pellets with refractory metal carbides for fission product containment, and no additional coating is necessary as with earlier graphite matrix and composite fuels.

Before any of the exotic advanced fuels mentioned in Section 3.2.2.3, or other advanced fuel candidates, could be seriously considered for development as a potential replacements for TRISO fuel in the VHTR, an evaluation would need to be performed to determine if the materials are neutronically, mechanically, and chemically compatible with the VHTR reactor design and with fuel performance requirements. The candidates that passed this analytical screening evaluation would require extensive irradiation and postirradiation heating testing before their suitability for use as VHTR fuel could be assured.

6.2.3 Compact Fabrication

For large-scale fuel manufacturing, a thermosetting-matrix-based compacting process (which includes matrix overcoating of the particles prior to compacting) is preferred over the reference U.S. thermoplastic-matrix-based process for a number of reasons. In principle, the thermosetting-matrix-based process should result in improved fuel quality because (1) thermosetting matrix is formulated using synthetic thermosetting resins that have much lower impurity levels than the petroleum pitch used as the binder in thermoplastic matrix, (2) the thermosetting matrix should yield stronger, less friable compacts, and (3) the thermosetting matrix process should require lower compacting forces and less handling of the compacts, thereby reducing the potential for damage. In addition, a thermosetting-matrix-based process

---

[17] http://www.cea.fr/gb/publications/Clefs45/clefs45gb/clefs4521a.html

[18] http://www.fas.org/nuke/space/c04rover.htm)

[19] http://www.inspi.ufl.edu/tricarbide.pdf)

would eliminate the need to pack the compacts in alumina powder during carbonization, and would therefore be better suited to automation, which would reduce the cost of fuel compact fabrication. Because of these inherent advantages, the AGR Program has chosen to develop a thermosetting-matrix-based process as the reference compact fabrication process for the VHTR fuel.

WBS 2.1.3: It is assumed that the AGR Program will be successful in developing a thermosetting-matrix-based compact fabrication process capable of producing fuel compacts that meet VHTR quality requirements, and that this process will be equally suitable for compacting ZrC-TRISO or standard TRISO particles. Consequently, little or no compacting development is required under this development plan beyond liaison with the AGR Program. However, it is conceivable that the TRIZO coatings would have a different susceptibility to damage during compact formation than conventional TRISO coatings. Tests will be performed to determine if the compact process conditions need to be optimized for TRIZO particles.

However, if ZrC-TRISO is selected as the reference VHTR fuel type, this may permit VHTR fuel compacts to be final heat treated at a substantially higher temperature than are compacts containing standard TRISO fuel particles. Heat treatment at a higher temperature should improve the graphitization of the matrix, which should increase the thermal conductivity of the fuel compacts thereby reducing fuel temperatures and should make the matrix more oxidation resistant. In this event, a limited amount of research and development would be needed to finalize the thermal conditions (e.g., peak temperature and temperature ramp rates) for final heat treatment.

The reference compact fabrication process developed by the AGR Program (perhaps with modifications to the formation and final heat treatment steps) will be used to fabricate compacts from UO_2* or ZrC-TRISO particles to support the irradiation testing program

6.2.4 Quality Control Techniques

WBS 2.1.4: It is assumed that the necessary QC methods development that is generic for TRISO-coated fuel (see Section 3.1.4) will be successfully completed by the AGR Program. Consequently, the QC methods development to be conducted under this plan is limited to new methods specifically needed for ZrC-TRISO or UO_2^* fuel particles.

As discussed in Section 3.1.4, ZrC does not form a protective oxide layer (as does SiC) when exposed to air at high temperature; consequently, several standard QC methods employed for SiC-TRISO fuel cannot be used for ZrC-TRISO fuel. This includes the burn-leach test for defective SiC and all QC methods involving "burn-back" of coated particles or coating fragments in air to segregate the SiC layer from pyrocarbon. The much higher density of ZrC (6.6 g/cm^3) relative to that of SiC (3.2 g/cm^3) is also problematic with respect to coating density measurements and the use of x-radiographic inspection for coating thickness, missing buffer coatings, and heavy metal dispersion.

Ogawa and Fukuda (Ogawa 1989) developed a plasma oxidation technique that is capable of completely removing pyrocarbon from ZrC without significant oxidation of the ZrC. Ogawa and Fukuda (Ogawa 1990) have also applied plasma oxidation with emission monitoring to the quantitative analysis of the free carbon content in ZrC coatings, a property which is very important to the fission product retentiveness of ZrC. Further development of these techniques will be conducted to formally qualify them as QC methods. In addition, QC methods specifically for Zr-C TRISO fuel will be developed for ZrC density, heavy metal dispersion, ZrC defect fraction, and missing buffer fraction.

Quality Control methods specific to UO_2* fuel may also need to be developed depending on the requirements specified for the fuel. New methods that may require development include:

- Methods to characterize the thickness and density of the seal coat between the UO_2 kernel and the ZrC coating;
- Methods to characterize the thickness, density, and C/Zr ratio of the ZrC coating in the UO_2*-C particles and the ZrC defect fraction;
- A method to measure the C/Zr content of the buffer layer in UO_2*-B particles;
- A method to measure ZrC defect fractions.

6.2.5 Test Fuel Fabrication

WBS 2.1.5: Test fuel specimens will be fabricated for the irradiation tests described in Section 6.2 using the processes developed under this program for the production of UO_2* and TRIZO-coated UCO (Sections 6.2.1 through 6.2.3). The test fuel specimens will be fabricated to product, process and equipment specifications prepared to and controlled by the QC requirements defined in Section 8. The test fuel will be qualified using standard QC protocols for TRISO fuels as well as specialized QC techniques developed under this program (Section 6.2.4).

6.2.6 Scrap Recovery and Recycle

WBS 2.1.6: The fundamentals of the processes to recover uranium from scrap TRISO fuel production materials are well established. Recovery of uranium from ZrC-TRISO fuel should actually be less difficult because the ZrC coatings can be fully oxidized, whereas this is not possible with SiC coatings. Consequently, no fundamental process issues that need to be addressed by a research and development program have been identified at this writing. Rather, the required development associated with uranium recovery and recycling is scale up of the existing processes for large-scale fuel manufacturing. This task a manufacturing issue that is beyond the scope of this development plan for advanced high temperature fuel. Consequently, the only workscope currently identified in this work area is liaison with the AGR Program. While not anticipated, it is conceivable that, as the fabrication of more exotic coating designs (e.g., NbC coatings, etc.) is explored, unique scrap recovery and recycle issues might be encountered that would be require additional workscope in this area.

6.3 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, during postirradiation examination, and during postirradiation heating tests.

6.3.1 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, 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.

6.3.2 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 (to provide the experimental basis for validating the fuel performance models).

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,[20] 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 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 5. A matrix of the planned capsule tests and the DDNs that they will address is given in Table 6-1. Summary descriptions of the capsules are given in Tables 6-2

---

[20] 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.

(DDNs addressed), 6-3 (test objectives) and 6-4 (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.

6.3.2.1 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 which will be fabricated in laboratory-scale equipment. 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 postirradiation 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 NbC, fission product getters in the kernel, etc.) which promise superior irradiation and accident performance at very high temperatures. The test fuel will be fabricated in laboratory-scale equipment. 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 9) 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.

6.3.2.2 Qualification Testing of Reference Fuel

WBS 2.2.2.2: Irradiation test **VHTR-3** is a test of reference fuel, produced in a large-diameter coater, to peak VHTR service conditions (e.g., 1400 °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, produced in a large-diameter coater, 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 fast 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 in a large-diameter coater 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 postirradiation heating tests which will include the effects of air and water ingress. It will also provide feedback to finalize product and process specifications.

6.3.2.3 Validation Testing of Reference Fuel

WBS 2.2.2.3: Irradiation test **VHTR-7** with reference fuel compacts, fabricated with optimized equipment including a large-diameter coater 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 and accident 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 5.5×10^{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 6-2 through 6-4 are fission product transport tests which are described in Section 6.2.)

6.3.3 Postirradiation Examination

WBS 2.2.3: The standard scope of the postirradiation examination for each of the irradiation capsules described in Section 6.2.2 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 6-5.

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 postirradiation 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 interfaces 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 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 Postirradiation Heating: Conduct postirradiation 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 9.

6.3.4 Postirradiation Heating Tests

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

6.3.4.1 Test Facility Construction

WBS 2.2.4.1:. The Core Conduction Cooldown Test Facility (CCCTF) is currently available at ORNL, and a new postirradiation heating facility is planned to be constructed on the AGR Program, perhaps at Argonne National Laboratory – West (ANL-W). Additional postirradiation 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 °C and heating in helium/air and helium/steams mixtures to at least 1400 °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 have 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.

6.3.4.2 Test Matrix

The planned postirradiation heating tests are summarized in Table 6-6, 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 °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 °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 °C temperature was chosen for continuity with the existing international postirradiation heating data base for conventional TRISO-coated particles, and the 2000 °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 °C to the earlier data at 2000 °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 °C will use an irradiated fuel compact that experienced severe irradiation conditions for comparison with the other 2000 °C heating data to determine the effects on accident performance. Heating test **PIH-14** at 2200 °C will use an irradiated fuel compact that experienced less severe irradiation conditions (cell 4) to determine the effects of high postirradiation 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 °C for comparison with the reference fuel at 2200 °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 °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 °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 postirradiation heating tests shown in Table 6-6 are fission product transport tests which are described in Section 6.2.)

6.4 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 5-1) 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 conventional TRISO fuels.

The work scope addressed in this plan **(WBS 2.3.1)** is limited primarily to radionuclide transport in kernels, particle coatings, and fuel-compact matrix. As indicated in Table 5-1, 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 herein.

Significant quantities of H-3 are produced in an HTGR as a result of activation of He-3 in the primary helium coolant, ternary fission, neutron activation of Li impurities, and burnout of control materials; however, the H-3 produced the core is expected to be largely retained in the core materials (>99%). Moreover, the core graphite is expected to be a major sink for the H-3 produced in the primary coolant by neutron activation of the He-3. Consequently, the transport properties of H-3 in the core materials, especially the H-3 sorptivity of core graphites, must be quantified as a function of irradiation and environmental conditions. Likewise, the permeation rates of tritium through the intermediate heat exchanger tube materials must be determined.

6.4.1 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.

6.4.1.1 RN Transport during Normal Operation

WBS 2.3.1.1.1: Capsule **VHTR-8** will be designed to characterize the fission product release from failed UO_2^* and TRIZO particles.[21] 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)[22] particles will be seeded into selected 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 6-4. 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.

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

---

[21] As for the Fuel Materials irradiations (Section 6.2), 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.

[22] "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.

behavior. (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 pure neon); it is anticipated that this will allow a 100-150 ^{o}C variation in fuel temperature from the nominal temperatures specified in Table 6-4. 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 postirradiation examination of capsule **VHTR-8** will include the tasks indicated in Table 6-5. 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 an isotope-by-isotope mass balance.

In addition to the PIE tasks called out in Table 6-4, the piggyback samples irradiated in capsule **VHTR-8** will be recovered and characterized. A number of them will undergo specialized examinations and postirradiation 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.

WBS 2.3.1.1.7: In addition to the differential data to be obtained from capsule **VHTR-8** for model development, independent integral radionuclide (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 6-4. 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 postirradiation examination of capsule **VHTR-9** will include the tasks indicated in Table 6-5. These plans will be reviewed when the PIE specification is prepared.

WBS 2.3.1.1.9: Measure the release of H-3 from failed and intact reference fuel particles as a function of temperature and, as appropriate, of irradiation and environmental conditions. Determine the retentivity of fuel element and core structural graphite as a function of temperature and, as appropriate, irradiation and environmental parameters. The experimental techniques and equipment for making these measurements are well established (Gainey 1976).

6.4.1.2 RN Transport under Accident Conditions

The new postirradiation 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 ^{o}C in dry helium and up to at least 1400 ^{o}C in helium/air or helium/steam environments. The furnace purge gas will provide for control of the atmosphere and a means for continuous monitoring of fission gas release. There will be provisions for removable and replaceable deposition surfaces (essentially a “cold finger”) to monitor the release of condensable radionuclides 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 ^{o}C; 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 ^{o}C 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

6.4.2 RN Transport in Primary Circuit

Most of the DDNs related to radionuclide transport in the primary coolant circuit are generic; hence, they should be addressed by the AGR program. There are several exceptions. The materials of construction used for the IHX in a VHTR will likely be different from the materials used for the turbine blades and recuperator in a direct-cycle plant. Consequently, the sorptivities and reentrainment characteristics of key radionuclides for IHX materials of construction will be determined as part of this program. In addition, as previously stated, the DDNs related to tritium production and transport are of particular interest to a VHTR for hydrogen production because the tritium produced in the primary coolant circuit can permeate through the heat exchangers and contaminate the hydrogen product (Gainey 1976). Consequently, tritium permeation rates through IHX tube materials will also be determined as part of this program.

6.4.2.1 RN Transport during Normal Operation

WBS 2.3.2.1.1: The deposition of Cs, Ag, I and Te on IHX structural metals will be characterized. The sorptivities of these nuclides will be measured as a function of temperature, partial pressure, surface state, and coolant chemistry; to the extent practical, these sorption data will be obtained at representative partial pressures to avoid the orders of magnitude extrapolations which are necessary with the present data base. Particular attention will be given to the effects dust on the deposition process. The diffusivities of silver and cesium in IHX structural metals will be determined with special attention to the effects of surface films. The test facilities developed on the AGR Program for making such measurements on turbine blades and recuperator materials of construction will be used for these measurements if available, or those test facilities will be replicated.

WBS 2.3.2.1.2: The permeation rate of tritium through IHX materials of construction will be determined as a function of temperature, H3 partial pressure, system pressure, coolant impurity concentrations and tube surface state. The effects of thermal cycling, which would occur as a result of reactor startup, shutdown, and load following will be simulated. The test techniques and equipment needed for making such measurements are well established (e.g., Yang 1977). The appropriate facilities will be constructed as part of this program since tritium permeation measurements are included in the current AGR Plan.

6.4.2.2 RN Transport under Accident Conditions

WBS 2.3.2.2.1: The fractional reentrainment ("liftoff") of the radiologically important radionuclides (I, Sr, Cs, Te and Ag) from IHX materials of construction will be measured as a function of the controlling system parameters. Test variables which will be investigated include shear ratio, absolute wall shear stress, blowdown duration, temperature, humidity, and surface oxidation state (other influential parameters may be identified in course of the testing program). The effects of dust on the reentrainment characteristics of deposited activity will be quantified. The test facilities developed on the AGR Program for making such measurements on turbine blades and recuperator materials of construction will be used for these measurements if available, or those test facilities will be replicated.

6.5 References for Section 6

[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.

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

DeVelasco, R. I., "MHTGR Fuel Compact Process Specification," GA Document 909548, Rev. 0, General Atomics, September 1987.

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

Gainey, B. W., "A Review of Tritium Behavior in HTGR Systems," GA-A13461, General Atomic, April 1976.

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**, 413-427, September, 1977.

Hanson, D. L., "Screening Tests for the Selection of VHTR Advanced Fuel," PC-000510, Rev. 0, General Atomics, December 2003.

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

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

Maneke, J. and D. L. Hanson, "Verification and Validation Plan for Radionuclide Control Methods," DOE-HTGR-88125, Rev. 1, General Atomics, November 1988.

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

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

Myers, B. F., “Fuel Design Data Manual,” GA Document 901866, Rev. F, General Atomics, August 1987.

Richards, M. B., “Design of Coated Particle Driver and Transmutation Fuels,” GA-224-0-RDE-000135, Rev. A, General Atomics, September 2002.

Yang, L, W. Baugh, and N. Baldwin, “Study of Tritium Permeation Tests on Peach Bottom Steam Generator Tubes,” GA-A14376, General Atomic, June 1977.

Table 6-1. 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 Postirradiation 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 6-2. Irradiation Tests to Satisfy VHTR DDNs

Capsule	Description[23]	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

---

[23] 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 6-3. 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, fabricated in lab-scale equipment, 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 postirradiation heating tests. Sufficient performance data to permit the selection of the reference VHTR fuel particle for further qualification testing.
VHTR-2	<u>Screening – TRIZO</u> Irradiation of particles with UCO kernels and a ZrC coating replacing the SiC layer (“TRIZO”), produced in large-diameter coater.	Irradiation data for TRIZO particles to determine if they meet VHTR fuel requirements, and generation of samples for postirradiation heating tests. Sufficient performance data to permit the selection of the reference VHTR fuel particle for further qualification testing.
VHTR-3	<u>Qualification – Reference Fuel</u> Irradiation of reference VHTR fuel (assumed either UO_2^* or TRIZO), produced in large-diameter coater 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, produced in large-diameter coater 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 and equipment, including large-diameter coater, 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 particle designs (e.g., “exotic” coatings, getters, etc.) which promise superior irradiation and	Sufficient irradiation and accident performance data to determine if more “exotic” particle designs (e.g., refractory coatings such as NbC, kernel getters, etc.) promise sufficiently superior performance at

Capsule	Description	Primary Objective/Expected Result
	accident performance at very high temperatures; test fuel fabricated in lab-scale equipment.	high temperature and/or under oxidizing conditions to merit further development.
VHTR-7	<u>Validation – Reference Fuel</u> Irradiation of reference VHTR fuel fabricated to final product and process specifications in large-diameter 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	<u>FP Transport – Particles</u> 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. Same driver fuel as VHTR-3.	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	<u>Validation – FP Release</u> Irradiation of fuel compacts seeded with missing-buffer reference particles under peak VHTR core conditions. Same driver fuel as VHTR-7.	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 6-4. Capsule Design Conditions

Capsule	Cell[24]	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	$<5.0 \times 10^{25}$	1000	<26	Yes
	2A/2B	UO_2^*- Zr buffer TRISO	Compacts	5.0×10^{25}	1350	26	No
	3A/3B	UO_2^*- ZrC overcoat TRISO	Compacts	5.0×10^{25}	1350	26	No
	4	UO_2^*- ZrC overcoat TRISO	Compacts Loose particles	$<5.0 \times 10^{25}$	1000	<26	Yes
VHTR-2	1A/1B	TRIZO UCO/ZrC	Compacts Loose particles	$<5.0 \times 10^{25}$	1000	<26	Yes
	2A/2B	TRIZO UCO/ZrC	Compacts	5.0×10^{25}	1350	26	No
	3	TRIZO UCO/ZrC	Compacts	5.0×10^{25}	1350	26	No
	4	TRIZO UCO/ZrC	Compacts Loose particles	$<5.0 \times 10^{25}$	1000	<26	Yes
VHTR-3	1	Reference Fuel (UO_2^* or TRIZO)	Compacts	$<5.0 \times 10^{25}$	800	<26	Yes
	2A/2B	Reference Fuel (UO_2* or TRIZO)	Compacts	5.0×10^{25}	1200	26	Yes

---

[24] “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[24]	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 x 10^{25}	1400	26	No
	4	Reference Fuel (UO_2* or TRIZO)	Compacts	<5.0 x 10^{25}	1000	<26	Yes
VHTR-4	1	Reference Fuel (UO_2* or TRIZO)	Compacts	<5.0 x 10^{25}	>1400	<26	Yes
	2A/2B	Reference Fuel (UO_2* or TRIZO)	Compacts	>>5.0 x 10^{25}	1350	>26	Yes
	3A/3B	Reference Fuel (UO_2* or TRIZO)	Compacts	>5.0 x 10^{25}	>1400	>26	No
	4	Reference Fuel (UO_2* or TRIZO)	Compacts	5.0 x 10^{25}	1250	26	Yes
VHTR-5	1	Reference Fuel (Optimized process)	Compacts	<5.0 x 10^{25}	800	<26	No
	2	Reference Fuel (Optimized process)	Compacts	5.0 x 10^{25}	1200	26	No
	3A/3B	Reference Fuel (Optimized process)	Compacts	5.0 x 10^{25}	1400	26	No
	4A/4B	Reference Fuel (Optimized process)	Compacts	<5.0 x 10^{25}	1000	<26	No
VHTR-6	1	Advanced Particles (Variant #1)	Compacts Loose particles	5.0 x 10^{25}	1350	26	Yes (Variant #3)

Capsule	Cell[24]	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	$>5.0 \times 10^{25}$	1500	>26	Yes (Variant #3)
	3A/3B	Advanced Particles (Variant #2)	Compacts Loose particles	$>5.0 \times 10^{25}$	1500	>26	Yes (Variant #4)
	4	Advanced Particles (Variant #2)	Compacts Loose particles	5.0×10^{25}	1350	26	Yes (Variant #4)
VHTR-7	1	Reference Fuel (Final specs)	Compacts	$<5.0 \times 10^{25}$	900	<26	No
	2A/2B	Reference Fuel (Final specs)	Compacts	5.5×10^{25}	1200	≥29	No
	3A/3B	Reference Fuel (Final specs)	Compacts	5.5×10^{25}	1400	≥29	No
	4	Reference Fuel (Final specs)	Compacts	$<5.0 \times 10^{25}$	1100	<26	No
VHTR-8[25]	1	Failed Particles UO_2*-C/TRISO	Compacts Loose particles	$<5.0 \times 10^{25}$	1000[26]	<26	Yes
	2A/2B	Failed Particles UO_2^*-C/TRISO	Compacts Loose particles	5.0×10^{25}	1400	26	Yes

---

[25] 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.

[26] 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[24]	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 x 10^{25}	1200	26	Yes
	4	Failed Particles UO_2^*-B/TRISO	Compacts Loose particles	<5.0 x 10^{25}	1200	<26	Yes
VHTR-9[25]	1	Reference Fuel (MB particles)	Seeded Compacts	<5.0 x 10^{25}	800[27]	<26	No
	2A/2B	Reference Fuel (MB particles)	Seeded Compacts	5.0 x 10^{25}	1200[27]	26	No
	3A/3B	Reference Fuel (MB particles)	Seeded Compacts	5.0 x 10^{25}	1400[27]	26	No
	4	Reference Fuel (MB particles)	Seeded Compacts	<5.0 x 10^{25}	1000[27]	<26	No

---

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

Table 6-5. 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	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	X	X	
PIE TASK-9	Fuel Compact Deconsolidation	X	X	X	X	X	X	X		
PIE TASK-10	IMGA	X	X	X	X	X	X	X		
PIE TASK-11	Fuel Metallography	X	X	X	X	X	X	X	X	X
PIE TASK-12	Fuel Particle SEM Failure Mechanism	X	X	X	X	X	X	X	X	
PIE TASK-13	SEM Examination of Fission Products in Kernels & Coatings	X	X	X					X	
PIE TASK-14	Fission Gas and CO/CO_2 Content of Particle	X	X	X	X		X			
PIE TASK-15	Properties of Irradiated Materials Specimens	X	X							
PIE TASK-16	Radionuclide Transport in Irradiated Specimens	X	X		X				X	X
PIE TASK-17	Fission Product Release During Postirradiation Annealing	X	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 6-6. Postirradiation Compact Heating Test Matrix

Test	Test Article (Burnup)[2]	Temperature (^{O}C)	Time History	Test Objectives[28]						
				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		

---

[28] 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

[2] ”(High)” denotes High burnup
“(Low)” denotes Lower burnup

Test	Test Article (Burnup)[2]	Temperature (^{O}C)	Time History	Test Objectives[28]						
				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[29]						X	

---

[29] "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)[2]	Temperature (^{O}C)	Time History	Test Objectives[28]						
				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

7. Facility Requirements for VHTR Fuel Development Plan

7.1 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. Not unsurprisingly then, 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. If the requisite facilities do not exist (e.g., new postirradiation heating facilities), they have been provided for in the cost estimates for this program. Specifically, the cost estimate includes monies for the construction of a large-diameter coater which will utilize the coater design being developed on the AGR program. In addition, two new postirradiation heating facilities will be designed and constructed. The point of departure for the design will be the upgraded CCCTF under development on the AGR program; however, the design will need to be modified to permit heating tests in dry helium to 2200 ^{o}C and heating tests with oxidizing atmospheres up to at least 1400 ^{o}C.

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

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

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.

7.2 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.

Fuel fabrication and characterization:

- Metals and Ceramics Research and Development Laboratory, Building 4508 (ORNL)
- Process Development Laboratory, Building 4501 (ORNL)
- Post Irradiation Examination Laboratory, Building 4501 (ORNL)
- UCO (and UO_2 precursor) kernel process line (BWXT)
- Two intermediate-diameter (6”) coaters (BWXT)

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)

7.2.1 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, VA.

The AGR Program has already begun the process of recommissioning kernel and coating facilities at ORNL and BWXT.

7.2.2 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 US gas reactor programs to irradiate coated-particle fuel. Two specific materials irradiation facilities are of note. 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. 7-1; a view of the reactor and its storage basin is shown in Fig. 7-2.

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 are 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. 7-3; a view of the reactor core and pool is shown in Fig. 7-4.

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 been used to irradiate GA compacts for the US program in the late 1980s. Two different types of irradiation rigs/locations are available: 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, which 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-1, 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 US 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 limit 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 the 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. Ampoules are currently being designed for the RBT-6 reactor the International GT-MHR program. These new facilities will permit multi-cell irradiations of loose particles and compacts; design details are not available at this writing. 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.

7.2.3 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 their status are summarized in Table 7-1. 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 of pyrocarbon coatings.

7.2.4 Post-Irradiation Accident Test Facilities

The Core Conduction Cooldown Test Facility 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. 7-5, 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.

7.3 Selected Test Facilities for Advanced Fuel Development

The facility requirements for this incremental 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.

7.3.1 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 may be replicated and dedicated to this program (e.g., for ZrC coating development). Specifically, the cost estimate includes monies for the construction of a large-diameter coater which will utilize the coater design being developed on the AGR program.

7.3.2 Facilities for Irradiation Testing of Advanced Fuel

As described in Section 6, 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 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.

7.3.3 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 postirradiation heating facility as discussed in the next section.

7.3.4 Post-Irradiation Accident Test Facilities

Two new postirradiation 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 6, 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 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 7-1. 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 7-1. Building Complex Housing HFIR, ORNL

Figure 7-2. HFIR Reactor viewed through pool, ORNL

Figure 7-3. Building, Complex Housing ATR, INEEL

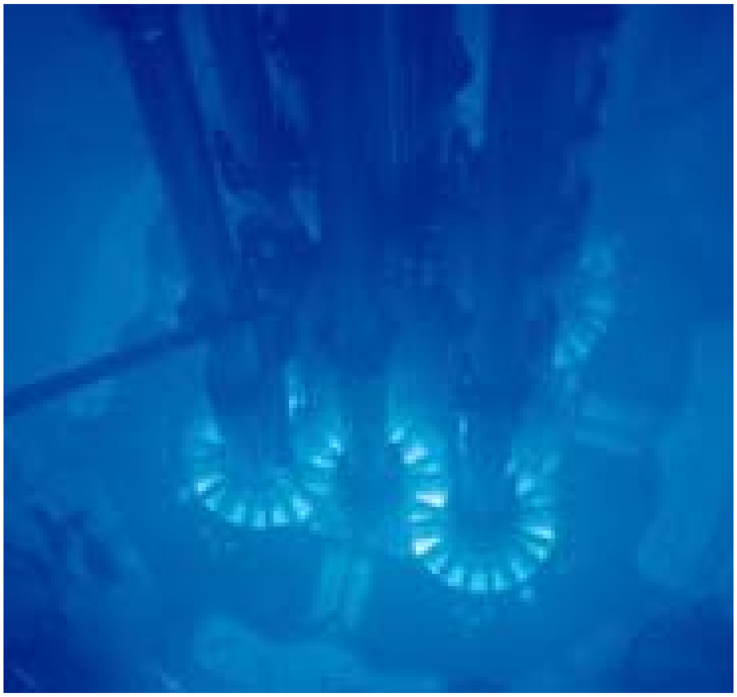


Figure 7-4. ATR Core and Pool, ORNL

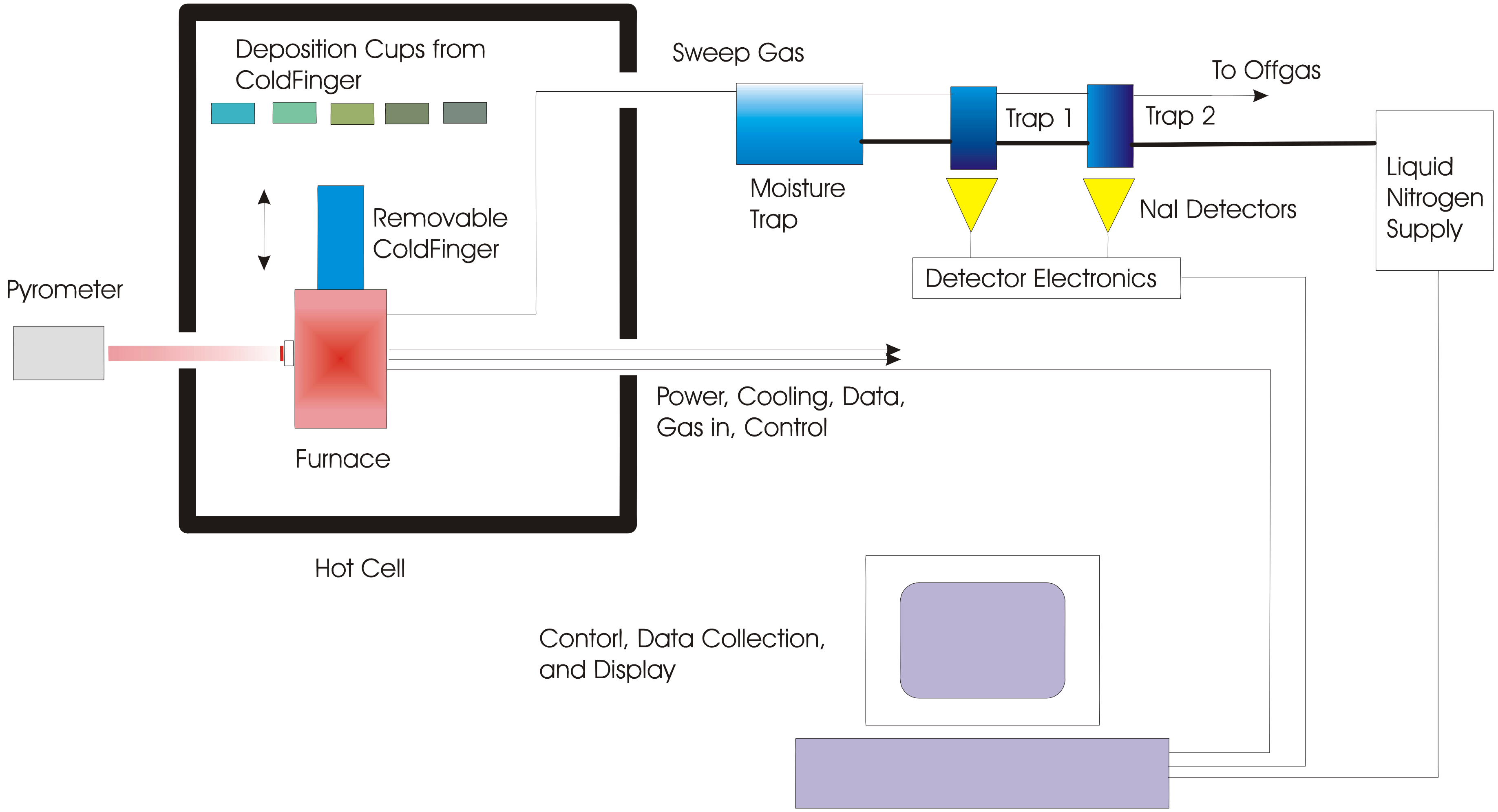


Figure 7-5. Schematic of CCCTF, ORNL

8. 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.

9. Schedule and Cost for the Program

The cost and schedule for the program are arranged in a work breakdown structure (Table 6-1) 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 9-1.

As noted previously, this umbrella development plan was preceded by a screening plan (PC-000510/0) which focused on the irradiation and accident simulation tests needed to select and qualify an advanced fuel for the VHTR. When the process development tasks were planned in more detail in the preparation of this umbrella, the scope, schedule and cost estimates changed somewhat from the earlier estimates in the screening plan (e.g., the total cost estimate increased by about $3 million). Consequently, the schedule and cost estimates presented herein supercede the earlier ones in the screening.

9.1 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). The durations of key tasks (e.g., capsule irradiation, postirradiation examination, postirradiation heating, etc.) were chosen to be consistent with the detailed estimates developed on the AGR Program. The planned program continues into FY2016 to complete postirradiation work on a planned screening capsule with more exotic coatings.

9.2 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.

The cost estimate includes monies for the construction of a large-diameter coater which will utilize the coater design being developed on the AGR program. In addition, two new postirradiation heating facilities will be designed and constructed. The point of departure for the design will be the upgraded CCCTF under development on the AGR program; however, the design will need to be modified to permit heating tests in dry helium to 2200 °C and heating tests with oxidizing atmospheres up to at least 1400 °C.

As summarized in Table 1-2, the total cost of the planned program is about $80 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 postirradiation 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 result in 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 9-1. Basis for Duration and Cost Estimate

WBS	Task Title	Duration (Days)	Cost Estimate	Cost Basis	Comments
1.	Fuel Design	4109	$3,296,939		Collector
1.1	Design Data Needs	1552	$225,053		Collector
1.1.1	Issue 0	92	$75,565	3 man-mo at $25K/man-mo	EJ[30]
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	$225,053		Collector
1.2.1	Issue 0	92	$75,565	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	1278	$375,362		Collector
1.3.1	Issue 0	92	$75,565	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
1.3.4	Technical Support Document	183	$150,309	6 man-mo at $25K/man-mo	EJ
1.4	Model Development	913	$597,950		Collector
1.4.1	Particle Performance	821	$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

---

[30] "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	Radionuclide Transport	821	$149,488		Collector
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
1.4.3	Fuel Design Data Manual	821	$298,975		Collector
1.4.3.1	Issue 1 (Preliminary Design)	91	$74,744	3 man-mo at $25K/man-mo	EJ
1.4.3.2	Issue 2 (Final Design)	91	$74,744	3 man-mo at $25K/man-mo	EJ
1.4.3.3	Technical Support Document	182	$149,488	6 man-mo at $25K/man-mo	
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	4474	$77,094,538		Collector
2.1	Fuel Process Development	3284	$14,676,668		Collector
2.1.1	Kernel Process Development	2556	$2,373,525		Collector
2.1.1.1	UCO Kernel Optimization	730	$59,959	0.1 eq hd/yr for 2 yr @ $300K/yr	EJ
2.1.1.2	UO_2^* Kernel Development	730	$59,959	0.1 eq hd/yr for 2 yr @ $300K/yr	EJ
2.1.1.3	Advanced Kernel Process Dev	1826	$149,980	0.1 eq hd/yr for 5 yr @ $300K/yr	EJ
2.1.1.4	Kernel Fabrication	181	$2,103,626	Cost for analogous AGR task	
2.1.2	Coating Process Development	2920	$6,862,750		Collector
2.1.2.1	TRISO Coating Process Opt.	730	$59,959	0.1 eq hd/yr for 2 yr @ $300K/yr	EJ
2.1.2.2	ZrC Coating Process Dev	2737	$5,303,809		Collector
2.1.2.2.1	Small Coater Studies	270	$1,370,521	6 eq hd for 9 mo + materials	EJ

WBS	Task Title	Duration (Days)	Cost Estimate	Cost Basis	Comments
2.1.2.2.2	Zr Feed System	120	\$499,715	5 eq hd for 4 mo + materials	EJ
2.1.2.2.3	Large Coater Construction	365	\$2,008,636	Cost for analogous AGR task	
2.1.2.2.4	Large Coater Studies	365	\$950,355	3 eq hd for 12 mo + materials	EJ
2.1.2.2.5	Process Optimization	180	\$474,582	3 eq hd for 6 mo + materials	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	731	\$444,307	0.75 eq hd/yr for 2 yr @ \$300K/yr	EJ
2.1.4	QC Development	2283	\$2,398,371		Collector
2.1.4.1	UO_2^*/TRIZO QC Methods	730	\$1,798,778	3 eq hd/yr for 2 yr @ \$300K/yr	EJ
2.1.4.2	Exotic Fuels QC Methods	730	\$599,593	1 eq hd/yr for 2 yr @ \$300K/yr	EJ
2.1.5	Test Fuel Fabrication	3010	\$2,537,756		Collector
2.1.5.1	Screening Tests	3010	\$1,109,575		Collector
2.1.5.1.1	VHTR-1 Capsule (UO_2^*)	180	\$369,612	Cost for AGR-2 fuel	
2.1.5.1.2	VHTR-2 Capsule (TRIZO)	180	\$369,612	Cost for AGR-2 fuel	UCO from AGR?
2.1.5.1.3	VHTR-6 Capsule (Adv. Particles)	270	\$370,351	Cost for AGR-2 fuel	Low?
2.1.5.2	Qualification Tests	819	\$739,224		Collector
2.1.5.2.1	VHTR-3 Capsule (Ref. Fuel)	180	\$184,806	Cost for AGR qualification capsules	
2.1.5.2.2	VHTR-4 Capsule (Ref. Fuel)	180	\$184,806	Cost for AGR qualification capsules	
2.1.5.2.3	VHTR-5 Capsule (Ref. Fuel)	180	\$369,612	Larger batches for better statistics	EJ
2.1.5.3	Validation Tests	180	\$369,612		Collector
2.1.5.3.1	VHTR-7 Capsule (Ref. Fuel)	180	\$369,612	Larger batches for better statistics	EJ
2.1.5.4	FP Transport Tests	2646	\$319,345		Collector

WBS	Task Title	Duration (Days)	Cost Estimate	Cost Basis	Comments
2.1.5.4.1	VHTR-8 Capsule (UO_2^*/TRIZO)	180	$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	$59,959	0.1 eq hd/yr for 2 years @ $300K/yr	EJ
2.2	Fuel Materials Development	4474	$47,425,524		Collector
2.2.1	Out-of-Pile Characterization	1553	$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	3563	$14,710,714		Collector
2.2.2.1	Screening Tests	3563	$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	1461	$5,917,981		Collector
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	Postirradiation Examination	3200	$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	

WBS	Task Title	Duration (Days)	Cost Estimate	Cost Basis	Comments
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	3520	$24,215,418		Collector
2.2.4.1	Facility Construction	2105	$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	
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	

WBS	Task Title	Duration (Days)	Cost Estimate	Cost Basis	Comments
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	
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	

WBS	Task Title	Duration (Days)	Cost Estimate	Cost Basis	Comments
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	4017	$14,992,346		Collector
2.3.1	Transport in Reactor Core	3287	$12,444,077		Collector
2.3.1.1	Normal Operation	3287	$8,501,220		Collector
2.3.1.1.1	VHTR-8 (UO_2^*/TRIZO) Irrad.	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	
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	2467	$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	

WBS	Task Title	Duration (Days)	Cost Estimate	Cost Basis	Comments
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	1826	$2,548,269		Collector
2.3.2.1	Normal Operation	1461	$1,948,677		Collector
2.3.2.1.1	RN Sorption on VHTR Alloys	1095	$1,349,084	1.5 eq hd/yr for 3 yr @ $300K/yr	EJ
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

10. Deliverables

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

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

Fabricated Items are the composite physical components and materials made to satisfy the conduct of the task. In this program they will be mostly irradiation capsules, fuel, and fuel items. They will be discarded after they have served their purpose and sufficient archival samples have been 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 10-1 Deliverables Identified

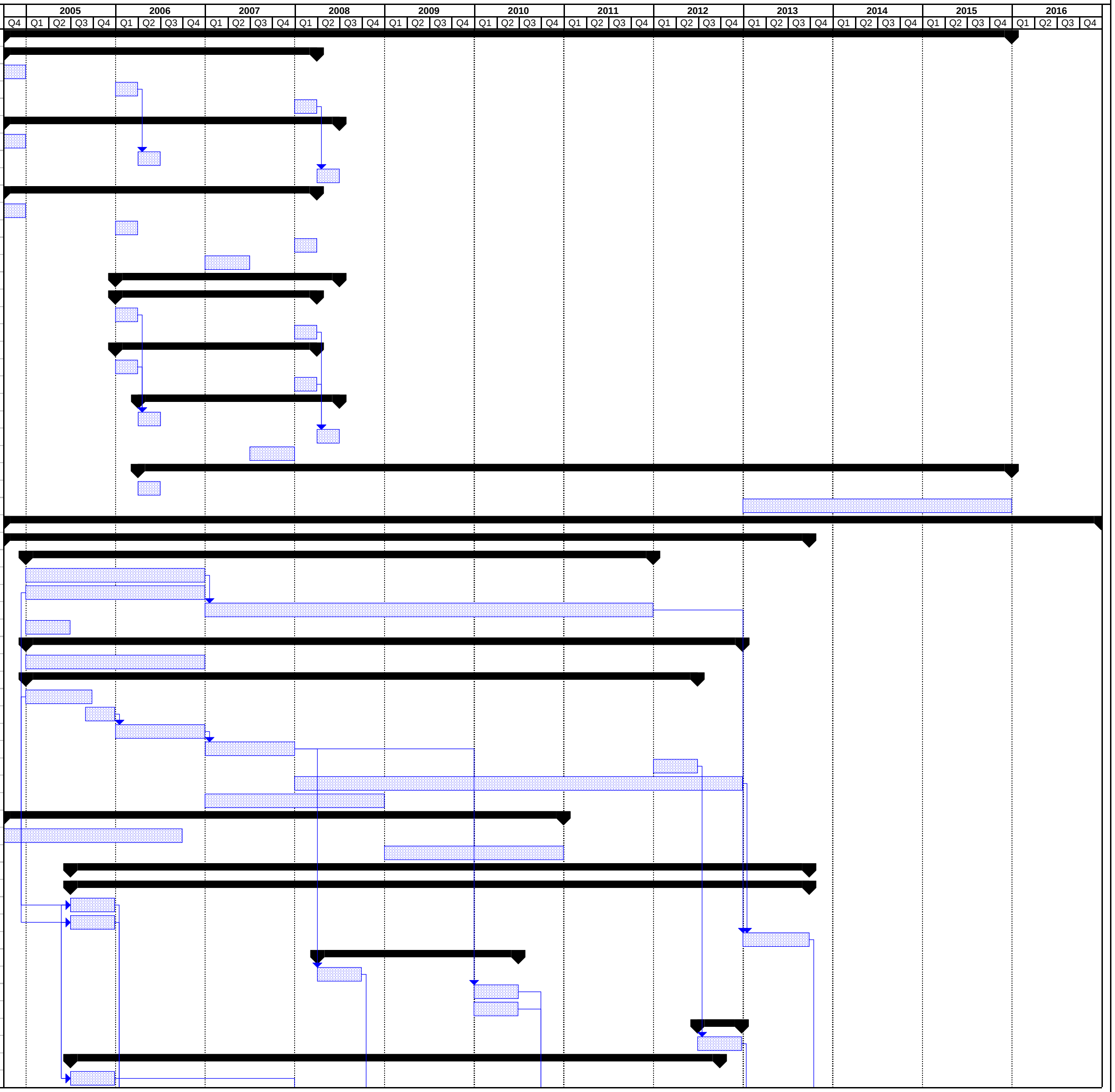
WBS	Task Name	Deliverable	Date
1.	Fuel Design	a. Fuel Development Plan (multiple issues) b. Fuel Specifications (multiple issues) c. Fuel Spec Technical Support Document d. FDDM (multiple issues) e. FDDM Technical Support Document f. Methods V&V Plan g. Methods Validation Report h. Letter reports on program status i. Quarterly progress reports j. Five-year status reports k. Final report l. QA Plan for the program m. 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)	At the completion of relevant progress, at completion of Task (See schedule, App.

WBS	Task Name	Deliverable	Date
		c. PIE specifications (seven capsules) d. Postirradiation heating specifications (seven capsules) 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. Postirradiation heating reports (seven capsules) l. Reference Fuel Selection Report	A)
2.3	Radionuclide Transport	a. Fission Product Chemical Forms and Implications Report b. Irradiation test specifications (two capsules) c. PIE specifications (two capsules) d. Postirradiation 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. Postirradiation 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**	**$3,296,939**
1.1	**Design Data Needs**	**$225,053**
1.1.1	Issue 0	$75,565
1.1.2	Issue 1 (Preliminary Design)	$74,744
1.1.3	Issue 2 (Final Design)	$74,744
1.2	**Fuel Development Plan**	**$225,053**
1.2.1	Issue 0	$75,565
1.2.2	Issue 1 (Preliminary Design)	$74,744
1.2.3	Issue 2 (Final Design)	$74,744
1.3	**Fuel Specifications**	**$375,362**
1.3.1	Issue 0	$75,565
1.3.2	Issue 1 (Preliminary Design)	$74,744
1.3.3	Issue 2 (Final Design)	$74,744
1.3.4	Technical Support Document	$150,309
1.4	**Model Development**	**$597,950**
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
1.4.3	**Fuel Design Data Manual**	**$298,975**
1.4.3.1	Issue 1 (Preliminary Design)	$74,744
1.4.3.2	Issue 2 (Final Design)	$74,744
1.4.3.3	Technical Support Document	$149,488
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**	**$77,094,538**
2.1	**Fuel Process Development**	**$14,676,668**
2.1.1	**Kernel Process Development**	**$2,373,525**
2.1.1.1	UCO Kernel Optimization	$59,959
2.1.1.2	UO2* Kernel Development	$59,959
2.1.1.3	Advanced Kernel Process Dev	$149,980
2.1.1.4	Kernel Fabrication	$2,103,626
2.1.2	**Coating Process Development**	**$6,862,750**
2.1.2.1	TRISO Coating Process Optimization	$59,959
2.1.2.2	**ZrC Coating Process Dev**	**$5,303,809**
2.1.2.2.1	Small Coater Studies	$1,370,521
2.1.2.2.2	Zr Feed System	$499,715
2.1.2.2.3	Large Coater Construction	$2,008,636
2.1.2.2.4	Large Coater Studies	$950,355
2.1.2.2.5	Process Optimization	$474,582
2.1.2.3	Nonconventional Coatings	$1,498,982
2.1.3	**Compact Process Development**	$444,307
2.1.4	**QC Development**	**$2,398,371**
2.1.4.1	UO2*/TRIZO QC Methods	$1,798,778
2.1.4.2	Exotic Fuels QC Methods	$599,593
2.1.5	**Test Fuel Fabrication**	**$2,537,756**
2.1.5.1	Screening Tests	**$1,109,575**
2.1.5.1.1	VHTR-1 Capsule (UO2*)	$369,612
2.1.5.1.2	VHTR-2 Capsule (TRIZO)	$369,612
2.1.5.1.3	VHTR-6 Capsule (Adv. Particles)	$370,351
2.1.5.2	Qualification Tests	**$739,224**
2.1.5.2.1	VHTR-3 Capsule (Ref. Fuel)	$184,806
2.1.5.2.2	VHTR-4 Capsule (Ref. Fuel)	$184,806
2.1.5.2.3	VHTR-5 Capsule (Ref. Fuel)	$369,612
2.1.5.3	Validation Tests	**$369,612**
2.1.5.3.1	VHTR-7 Capsule (Ref. Fuel)	$369,612
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.MP
Date: Tue 1/20/04

Task	▨	Milestone	◆	Rolled Up Task	▨	Rolled Up Progress	■	External Tasks	■	External Milestone	◆
Progress	■	Summary	■	Rolled Up Milestone	◇	Split	⋯	Project Summary	■	Deadline	⇩

Table A-1. Advanced Fuel Development Schedule for VHTR

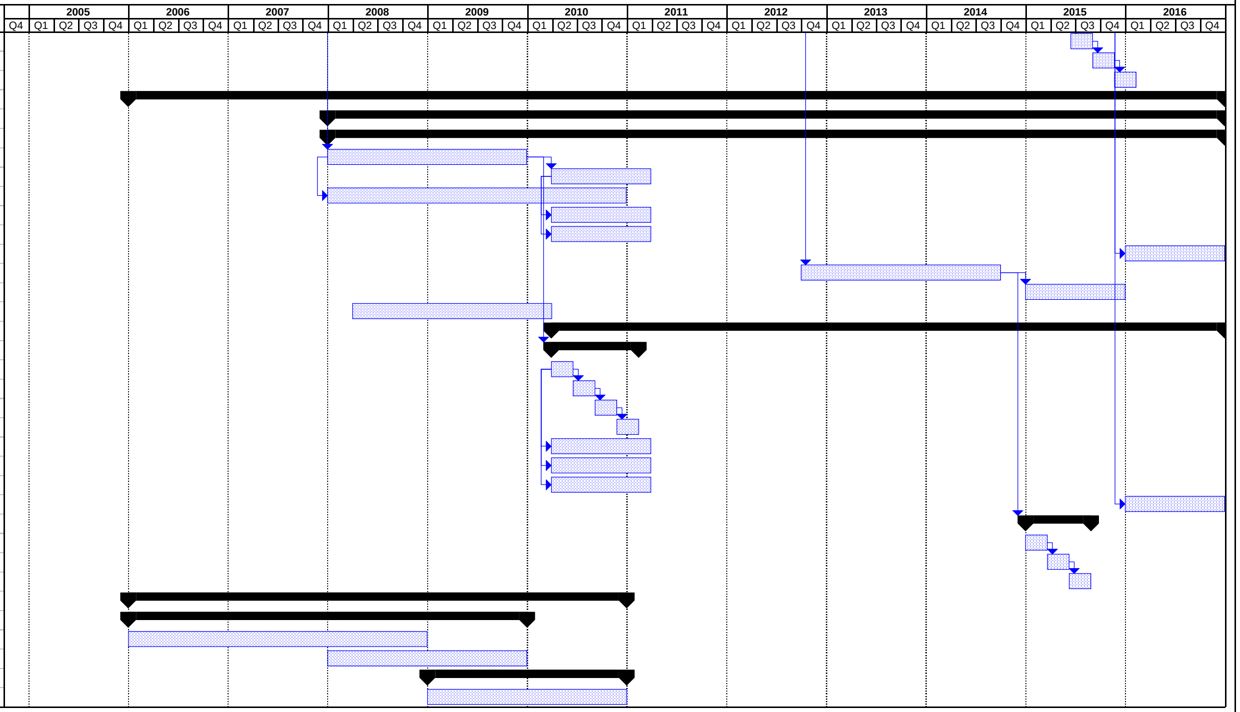
WBS	Task Name	Total Cost
2.1.5.4.2	VHTR-9 Capsule (Ref. Fuel)	$159,672
2.1.6	**Product Recovery Development**	$59,959
2.2	**Fuel Materials Development**	**$47,425,524**
2.2.1	**Out-of-Pile Characterization**	**$299,961**
2.2.1.1	Thermochemical Analysis	$119,919
2.2.1.2	Material Property Measurements	$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**
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

Project: Advanced Fuel Dev_Rev2.MP
Date: Tue 1/20/04

Task ▨
Progress ■
Milestone ◆
Summary ■
Rolled Up Task ▨
Rolled Up Milestone ◇
Rolled Up Progress ■
Split ┈
External Tasks ▨
Project Summary ■
External Milestone ◆
Deadline ⇩

Table A-1. Advanced Fuel Development Schedule for VHTR

WBS	Task Name	Total Cost
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 Coating	$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 Coating	$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.MP
Date: Tue 1/20/04

Task	▨	Milestone		Rolled Up Task	▨	Rolled Up Progress	▬	External Tasks	▬	External Milestone
Progress	▬	Summary	▬	Rolled Up Milestone		Split	....	Project Summary	▬	Deadline

Appendix B: Detailed Cost Estimate

Table B-1. Cost Estimate for Advanced Fuel Development

	2004	2005	2006	2007	2008	2009	2010	2011	2012	2013	2014	2015	2016	Total
Fuel Design														
Design Data Needs														
Issue 0	$75,565													$75,565
Issue 1 (Preliminary Design)			$74,744											$74,744
Issue 2 (Final Design)					$74,744									$74,744
Fuel Development Plan														
Issue 0	$75,565													$75,565
Issue 1 (Preliminary Design)			$74,744											$74,744
Issue 2 (Final Design)					$74,744									$74,744
Fuel Specifications														
Issue 0	$75,565													$75,565
Issue 1 (Preliminary Design)			$74,744											$74,744
Issue 2 (Final Design)					$74,744									$74,744
Technical Support Document				$150,309										$150,309
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
Fuel Design Data Manual														
Issue 1 (Preliminary Design)			$74,744											$74,744
Issue 2 (Final Design)					$74,744									$74,744
Technical Support Document				$149,488										$149,488
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		$29,980	$29,980											$59,959
UO2* Kernel Development		$29,980	$29,980											$59,959
Advanced Kernel Process Dev				$29,980	$30,062	$29,980	$29,980	$29,980						$149,980
Kernel Fabrication		$2,103,626												$2,103,626
Coating Process Development														
TRISO Coating Process Optimization		$29,980	$29,980											$59,959
ZrC Coating Process Dev														
Small Coater Studies		$1,370,521												$1,370,521
Zr Feed System		$499,715												$499,715
Large Coater Construction			$2,008,636											$2,008,636
Large Coater Studies				$947,751	$2,604									$950,355
Process Optimization									$474,582					$474,582
Nonconventional Coatings					$300,618	$299,796	$299,796	$299,796	$298,975					$1,498,982
Compact Process Development				$221,849	$222,457									$444,307
QC Development														
UO2*/TRIZO QC Methods	$226,695	$899,389	$672,694											$1,798,778
Exotic Fuels QC Methods						$299,796	$299,796							$599,593
Test Fuel Fabrication														
Screening Tests														
VHTR-1 Capsule (UO2*)		$369,612												$369,612
VHTR-2 Capsule (TRIZO)		$369,612												$369,612
VHTR-6 Capsule (Adv. Particles)										$370,351				$370,351
Qualification Tests														
VHTR-3 Capsule (Ref. Fuel)					$184,806									$184,806
VHTR-4 Capsule (Ref. Fuel)							$184,806							$184,806
VHTR-5 Capsule (Ref. Fuel)							$369,612							$369,612
Validation Tests														
VHTR-7 Capsule (Ref. Fuel)									$369,612					$369,612
FP Transport Tests														
VHTR-8 Capsule (UO2*/TRIZO)		$159,672												$159,672
VHTR-9 Capsule (Ref. Fuel)									$159,672					$159,672
Product Recovery Development					$30,062	$29,898								$59,959
Fuel Materials Development														
Out-of-Pile Characterization														
Thermochemical Analysis	$15,113	$59,959	$44,846											$119,919
Material Property Measurements			$59,959	$59,959	$60,124									$180,042

Table B-1. Cost Estimate for Advanced Fuel Development

	2004	2005	2006	2007	2008	2009	2010	2011	2012	2013	2014	2015	2016	Total
Irradiation Testing														
Screening Tests														
VHTR-1 Capsule (UO2*)		$9,339	$1,136,228	$1,126,890										$2,272,457
VHTR-2 Capsule (TRIZO)		$9,339	$1,136,228	$1,126,890										$2,272,457
VHTR-6 Capsule (Adv. Particles)										$286,392	$1,136,228	$849,837		$2,272,457
Qualification Tests														
VHTR-3 Capsule (Ref. Fuel)					$251,312	$986,330	$735,019							$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)									$13,511	$986,330	$975,521			$1,975,363
Postirradiation Examination														
VHTR-1 Capsule (UO2*)					$889,139	$274,071								$1,163,210
VHTR-2 Capsule (TRIZO)					$889,139	$274,071								$1,163,210
VHTR-3 Capsule (Ref. Fuel)							$9,561	$1,153,649						$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)												$932,679	$287,492	$1,220,171
Accident Simulation Tests														
Facility Construction														
PIH Furnace #1				$3,046,030	$953,254									$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					$262,991	$276,478								$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							$20,230	$519,239						$539,469
PIH-8								$539,469						$539,469
PIH-9								$539,469						$539,469
PIH-10								$539,469						$539,469
PIH-11								$323,682	$215,788					$539,469
PIH-12									$539,469					$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												$373,826	$392,996	$766,822
Radionuclide Transport														
Transport in Reactor Core														
Normal Operation														
VHTR-8 (UO2*/TRIZO) Irradiation					$1,139,341	$1,133,115								$2,272,457
VHTR-8 PIE							$1,006,527	$324,569						$1,331,096
FGR from UO2* Kernels					$135,252	$134,225	$134,539							$404,015
FM Diffusivities in UO2*							$188,157	$60,674						$248,831
FM Diffusivities in ZrC							$170,022	$54,826						$224,847
FM Diffusivities in Refractory Coatings													$224,847	$224,847

Table B-1. Cost Estimate for Advanced Fuel Development

	2004	2005	2006	2007	2008	2009	2010	2011	2012	2013	2014	2015	2016	Total
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							$208,757	$255,147						$463,904
FGR from UO2* Kernels							$188,157	$60,674						$248,831
FM Diffusivities in UO2*							$188,157	$60,674						$248,831
FM Diffusivities in ZrC							$74,810	$24,123						$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	$468,504	$5,940,724	$6,121,432	$7,308,839	$8,280,046	$5,768,491	$6,334,024	$6,758,101	$7,794,945	$8,016,745	$5,306,191	$7,968,079	$4,325,356	$80,391,477