WSRC-TR-98-00228

Bases for Functional Performance Requirements for a Spent Nuclear Fuel Treatment and Storage Facility

Savannah River Technology Center
Strategic Materials Technology Department
Materials Technology Section

Publication Date: July 1998

UNCLASSIFIED
DOES NOT CONTAIN
UNCLASSIFIED CONTROLLED
NUCLEAR INFORMATION

ADC & Reviewing Official: C.E. Sessions

Date: 6/25/98

Patent Status
This internal management report is being transmitted without DOE patent clearance, and no further dissemination or publication shall be made of the report without prior approval of the DOE-SR patent counsel.

Westinghouse Savannah River Company
Savannah River Site
Aiken, SC 29808

This document was prepared in connection with work done under Contract No. DE-AC09-96SR18500 with the U. S. Department of Energy

9903110159 990301
PDR WASTE
WM-1 PDR

109.7

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DOCUMENT: WSRC-TR-98-00228

TITLE: BASES FOR FUNCTIONAL PERFORMANCE REQUIREMENTS FOR A SPENT NUCLEAR FUEL TREATMENT AND STORAGE FACILITY

TASKS: SRT-MTS-97-2030

APPROVALS

Date: 6-25-98

T. M. Adams, Editor
SRTC-MATERIALS TECHNOLOGY SECTION

Date: 6-25-98

R. L. Sindelar, Editor
SRTC-MATERIALS TECHNOLOGY SECTION

Date: 6-25-98

D. W. Vinson, Editor
SRTC-MATERIALS TECHNOLOGY SECTION

Date: 6-25-98

P. E. Zapp, Technical Editor
SRTC-MATERIALS TECHNOLOGY SECTION

Date: 6-25-98

N. C. Iyer, SRTC-SNF Program Manager
Materials Applications & Corrosion Technology Group
SRTC-MATERIALS TECHNOLOGY SECTION

Date: 6/25/98

T. L. Capeletti, Manager
SRTC-MATERIALS TECHNOLOGY SECTION

Date: 6/26/98

M. W. Barlow, Manager
Advanced Programs
SPENT FUEL STORAGE DIVISION

CONTRIBUTORS TO WSRC-TR-98-00228

ATP Technology Development Task Leads (alphabetical order)
H. B. Peacock, SRTC
R. L. Sindelar, SRTC
B. J. Wiersma, SRTC

Editors (alphabetical order)
T. M. Adams, SRTC
R. L. Sindelar, SRTC
D. W. Vinson, SRTC

Contributors (alphabetical order)
T. M. Adams, SRTC
J. Brooke, SRTC
G. R. Caskey, Jr., SRTC
M. Dalmaso, SRTC
P. J. French, SRTC
M. H. Hodges, SRTC
J. P. Howell, SRTC
M. L. Hyder, SRTC
N. C. Iyer, SRTC
J. D. Jones, SRTC
P. S. Lam, SRTC
W. S. Large, SFSD
D. R. Leader, SRTC
S. Y. Lee, SRTC
M. R. Louthan, Jr., SRTC
J. I. Mickalonis, SRTC
H. B. Peacock, Jr., SRTC
F. C. Rhode, SFSD
R. L. Sindelar, SRTC
D. W. Vinson, SRTC
B. J. Wiersma, SRTC

List of Acronyms

Acronym	Definition
Al-SNF	Aluminum-clad, Aluminum-based Spent Nuclear Fuel
ANS	American Nuclear Society
ATR	Advanced Test Reactor
ATP	Alternative Al-SNF Treatment Technology Program
BOL	Beginning of Life
CFR	Code of Federal Regulations
CRWMS	Civilian Radioactive Waste Management System
DD	Direct Disposal
DIS	Disposability Interface Specification
DOE	U.S. Department of Energy
DOE-EM	U.S. Department of Energy Office of Environmental Management
DOE-RW	U.S. Department of Energy Office of Civilian Radioactive Waste Management (also OCRWM)
DRR	Domestic Research Reactor
DWPF	Defense Waste Processing Facility
EIS	Environmental Impact Statement
EOL	End of Life
FPR	Functional Performance Requirements
FRR	Foreign Research Reactor
HEPA	High-Efficiency Particulate Air
HEU	Highly-Enriched Uranium (> 20% ^{235}U in U)
HFBR	High-Flux Beam Reactor
HFIR	High Flux Isotope Reactor
HLW	High Level Waste
IAEA	International Atomic Energy Agency
ICD	Interface Control Document
INEL	Idaho National Engineering Laboratory
LEU	Low-Enriched Uranium (2 to 20% ^{235}U in U)
MD	Melt-Dilute
MGDS	Mined Geologic Disposal System
MIT	Massachusetts Institute of Technology
MTHM	Metric Tons Heavy Metal
MTR	Materials Test Reactor
MTRE	Materials Test Reactor Equivalent
MWd	Megawatt·days
NDE	Non-Destructive Examination
NRC	U.S. Nuclear Regulatory Commission
OCRWM	The Office of Civilian Radioactive Waste Management
ORNL	Oak Ridge National Laboratory
ORR	Oak Ridge Research Reactor
RHF	Réacteur à Haut Flux
RT	Room Temperature
RW	Office of Civilian Radioactive Waste Management
SFSD	Spent Fuel Storage Division
SNF	Spent Nuclear Fuel
SRS	Savannah River Site
SRTC	Savannah River Technology Center
TBD	To Be Determined or Technical Bases Document
TBV	To Be Verified
TSF	Treatment and Storage Facility

Table of Contents

1.0 EXECUTIVE SUMMARY 1.1

2.0 INTRODUCTION 2.1

2.1 BACKGROUND 2.1
2.2 EVOLUTION OF THE ALTERNATE TECHNOLOGY PROGRAM AND TECHNICAL STRATEGY FOR DISPOSITION OF AL-SNF 2.1
2.3 REPORT ORGANIZATION 2.2
2.4 REFERENCES 2.2

3.0 CROSS-REFERENCE TO FPR 3.1

4.0 MELT-DILUTE PROCESS STEPS AND FPRs 4.1

4.1 MELT-DILUTE TREATMENT OVERVIEW 4.1
4.2 PRE-TREATMENT DRYING 4.3
4.2.1 Background *4.3*
4.2.2 Preventing Melt-Water Interactions *4.3*
4.2.3 Pre-Treatment Furnace Drying *4.4*
4.3 PRE-TREATMENT CHARACTERIZATION 4.4
4.3.1 Melt-Dilution *4.5*
4.3.1.1 Background 4.5
4.3.1.2 In-Process Characterization Requirements 4.5
4.4 AL-SNF TREATMENT AND PRODUCT CONSISTENCY 4.6
4.4.1 Candidate Uranium Alloy Compositions *4.6*
4.4.1.1 U-Al SNF 4.6
4.4.1.2 U-Al-Si SNF 4.7
4.4.1.3 U_3O_8-Al SNF 4.9
4.4.1.4 Sterling Forest Oxide Type SNF 4.9
4.4.2 Process Cycle Options *4.9*
4.4.2.1 U-Al SNF 4.10
4.4.2.2 U-Al-Si SNF 4.11
4.4.2.3 U_3O_8-Al SNF 4.12
4.4.2.4 Sterling Forest Oxide-Type Material 4.13
4.4.3 Melt-Dilute Casting Techniques *4.13*
4.4.3.1 Top Pouring 4.13
4.4.3.2 Bottom Pouring 4.13
4.4.3.3 In-Situ Solidification 4.13
4.4.3.4 Casting Technique Requirements/Recommendations 4.14
4.4.4 Crucible Materials and Batch Size Requirements. *4.14*
4.4.4.1 Temperature 4.14
4.4.4.2 Melt Interactions 4.14
4.4.4.3 Batch Size 4.15
4.5 POST-TREATMENT CHARACTERIZATION 4.16
4.6 FURNACE REQUIREMENTS 4.17
4.6.1 Furnace Type *4.17*
4.6.1.1 Power and Cooling 4.18
4.6.1.2 Melting Operations 4.18
4.6.2 Stirring *4.19*
4.6.3 Furnace Power *4.19*
4.6.4 Atmosphere *4.19*
4.7 OFFGAS SYSTEM 4.19
4.7.1 Background and Overview *4.19*
4.7.2 Radionuclide Inventory for Research Reactor Fuel *4.20*
4.7.3 Offgas System Requirements *4.21*
4.7.3.1 Radionuclide Species of Interest 4.21
4.7.3.2 Temperature and Atmosphere Requirements 4.22

4.7.4 Offgas System Design *4.22*
4.7.4.1 Filter Media 4.22
4.7.4.1.1 Cesium Condenser 4.23
4.7.4.1.2 Iodine Reactor (if needed) 4.27
4.7.4.1.3 Krypton Absorber (if needed) 4.27
4.7.4.1.4 HEPA Filter 4.28
4.7.4.2 Flow Rates and System Pressure 4.28
4.7.4.3 Monitoring Filter Capacity 4.28
4.7.4.4 Stack Emissions 4.28
4.8 DISPOSITION OF SECONDARY WASTE 4.28
4.9 PACKAGING 4.30
4.9.1 Identification and Quantification of Materials Used as Neutron Absorbers *4.30*
4.9.2 Backfill Gas *4.31*
4.9.3 Canister Design *4.31*
4.9.4 Canister Labeling *4.32*
4.10 ROAD-READY STORAGE 4.32
4.10.1 Interim Dry Storage Requirements *4.32*
4.10.2 Monitoring During Canister Storage *4.33*
4.11 REFERENCES 4.33

5.0 DIRECT DISPOSAL PROCESS STEPS AND FPRs 5.1

5.1 AL-SNF RECEIPTS AT SRS—SHIPPING CASK HANDLING, AL-SNF HANDLING, AND CASK DECONTAMINATION FUNCTIONS 5.1
5.2 PRE-DRYING STORAGE 5.1
5.2.1 Pre-Drying Storage Requirements *5.2*
5.2.2 Basin Storage *5.2*
5.2.3 Open-Air Storage *5.3*
5.3 PRE-DRYING CHARACTERIZATION 5.4
5.3.1 Pre-Drying Characterization Requirements *5.4*
5.3.1.1 Background 5.4
5.3.1.2 Characterization Requirements for Al-SNF for Direct Disposal 5.5
5.3.2 Isotope Inventory of Fuel Elements and Assemblies *5.5*
5.3.3 Assembly Dimension Verification *5.5*
5.4 SNF DRYING TREATMENT 5.6
5.4.1 Drying Requirements *5.6*
5.4.2 Dryness Specification *5.7*
5.4.3 Vacuum Drying Specification *5.8*
5.5 POST-TREATMENT CHARACTERIZATION 5.8
5.6 PACKAGING 5.8
5.6.1 Identification and Quantification of Materials Used as Neutron Absorbers *5.8*
5.6.2 Backfill Gas *5.9*
5.6.3 Canister Design *5.9*
5.6.4 Canister Labeling *5.10*
5.7 ROAD-READY STORAGE 5.10
5.8 REFERENCES 5.11

A.0 SCOPE OF FUEL AT SRS TO BE DISPOSITIONED A.1

A.1 ALUMINUM-BASED SPENT NUCLEAR FUEL A.1
A.2 DOE SNF REPOSITORY PERFORMANCE CATEGORIES A.1
A.3 SRS RECEIPTS—AL-SNF A.1
A.3.1 Research Reactor Spent Nuclear Fuel Designs (Al-clad SNF Only) *A.1*
A.3.1.1 Plate-Type Design A.1
A.3.1.2 Concentric Tube Design A.4
A.3.1.3 Pin-Type Design A.4
A.3.1.4 Involute Type Design A.5
A.3.1.5 General Purpose Tubes A.8
A.3.2 U Oxide/Failed Clad & Al *A.9*

A.3.3 UAl_x/Al *A.10*
A.3.4 U-Si *A.15*
A.4 DESCRIPTION OF AL-SNF MATERIALS A.16
A.4.1 U Oxide/Failed Clad & Al *A.17*
A.4.2 UAl_x/Al *A.18*
A.4.3 U-Si *A.18*
A.4.4 Sterling Forest Oxide-Type Material *A.20*
A.4.5 Physical Condition of Fuel Assemblies *A.20*
A.4.6 Constituents of DOE SNF *A.20*
A.5 REFERENCES A.34

List of Tables

TABLE 3.1 CROSS-REFERENCE BETWEEN FUNCTIONAL PERFORMANCE REQUIREMENTS DOCUMENT AND TECHNICAL BASES DOCUMENT (TBD) 3.1
TABLE 4.1 LIST OF 39 ISOTOPES TO BE CHARACTERIZED AS PER THE DIS 4.16
TABLE 4.2 RADIONUCLIDES IN RESEARCH REACTOR SPENT FUEL 4.20
TABLE 4.3 ELEMENTS WITH BOILING POINTS BELOW 1000°C 4.22
TABLE 4.4 PARAMETERS TO PRECLUDE CRITICALITY FOR THE MD SNF FORM WITHOUT POISONS 4.31
TABLE 5.1 TABLE OF BASIN CHEMISTRY STANDARDS AND PRESENT CONDITIONS 5.3
TABLE 5.2 LIST OF 39 ISOTOPES TO BE CHARACTERIZED AS PER THE DIS 5.5

TABLE A.1 URANIUM OXIDE INVENTORY A.9
TABLE A.2 MIXED OXIDE INVENTORY A.10
TABLE A.3 URANIUM METAL INVENTORY A.10
TABLE A.4 URANIUM/ALUMINUM INVENTORY A.11
TABLE A.5 URANIUM/SILICON INVENTORY A.15
TABLE A.6 COMPOSITIONS OF TYPICAL MTR CLADDING ALLOYS MANUFACTURED BY CERCA, NUKEM, AND B&W A.21
TABLE A.7 COMPOSITION OF AL POWDER USED FOR UAL_x FUEL MANUFACTURING A.21
TABLE A.8 CHEMICAL ANALYSIS OF THE DEPLETED URANIUM FOR DILUTION A.21
TABLE A.9 BOUNDING ESTIMATES OF MASS OF FISSION PRODUCTS, ACTINIDES, AND LIGHT ELEMENTS (GRAMS) A.22
TABLE A.10 VAPOR PRESSURE IN ATMOSPHERES A.32
TABLE A.11 COMPOUNDS AND ALLOYS OF VARIOUS OFFGAS SPECIES WITH URANIUM AND ALUMINUM A.33

List of Figures

FIGURE 4.1 PROCESS STEPS IN TSF FOR MELT-DILUTE TECHNOLOGY 4.1
FIGURE 4.2 BINARY PHASE DIAGRAM OF THE URANIUM-ALUMINUM SYSTEM 4.2
FIGURE 4.3 TOTAL WASTE VOLUME FOR FRR, DRR, HFIR, AND ATR FUEL ASSEMBLIES AS A FUNCTION OF THE DILUTION FACTOR, PROCESS ADDITIONS AND WASTE COMPOSITION 4.3
FIGURE 4.4 POST-IRRADIATION MICROSTRUCTURE OF A U-AL-SI LEU FUEL 4.8
FIGURE 4.5 TERNARY ISOTHERMAL SECTIONS FOR THE U-AL-SI ALLOY SYSTEM 4.8
FIGURE 4.6 POST IRRADIATION MICROSTRUCTURE FOR A U_3O_8-AL HEU FUEL 4.9
FIGURE 4.7 SCHEMATIC TREATMENT CYCLE FOR THE MELT-DILUTE TREATMENT PROCESS 4.10
FIGURE 4.8 FULLY REACTED U_3Si_2 POWDER IN MOLTEN AL AT 850°C IN 30 MINUTES 4.11
FIGURE 4.9 EXPERIMENTAL DATA AND JOHNSON-MEHL-AVRAMI MODEL PREDICTIONS OF OXIDE DISSOLUTION IN MOLTEN ALUMINUM 4.12
FIGURE 4.10 CONCENTRATION OF IRON IN U-AL ALLOY MELTS FOLLOWING MELTING IN CARBON STEEL CRUCIBLES 4.15
FIGURE 4.11 SMALL SCALE RESISTANCE FURNACE APPARATUS 4.17
FIGURE 4.12 ESTIMATED CS-137 FOR FRR MTR INVENTORY 10 YEARS AFTER DISCHARGE 4.23
FIGURE 4.13 SCHEMATIC OF THE BENCH-SCALE OFFGAS SYSTEM APPARATUS 4.24
FIGURE 4.14 CESIUM RELEASE FROM BENCH-SCALE CARBON STEEL CRUCIBLE OFFGAS MELTS 4.25
FIGURE 4.15 XRF ANALYSIS OF AL-BILLET WITH BOROSILICATE GLASS FLUX 4.25
FIGURE 4.16 CESIUM RELEASE TO THE OFFGAS FILTER IN BENCH-SCALE OFFGAS EXPERIMENTS 4.26
FIGURE 4.17 SMALL-SCALE OFFGAS SYSTEM PHOTOGRAPH 4.26
FIGURE 4.18 SMALL-SCALE OFFGAS SYSTEM SCHEMATIC 4.27
FIGURE 4.19 SURROGATE MTR 4.27
FIGURE 4.20 FLOW SHEET OF MELT-DILUTE WASTE DISPOSAL TREATMENT OPTIONS 4.29
FIGURE 4.21 DRY OPTION FOR CESIUM LOADED FILTER BED 4.30
FIGURE 5.1 PROCESS STEPS IN TSF FOR DIRECT DISPOSAL TECHNOLOGY 5.2
FIGURE 5.2 INSTRUMENTED, SHIELDED TEST CANISTERS FOR VALIDATION TESTING 5.11

FIGURE A.1 TYPICAL (BOXED-TYPE/FLAT-PLATE) ALUMINUM-BASED FUEL ELEMENT SCHEMATIC A.2
FIGURE A.2 TYPICAL (BOXED-TYPE/CURVED-PLATE) ALUMINUM-BASED FUEL ELEMENT SCHEMATIC A.3
FIGURE A.3 TYPICAL MTR (TUBE-TYPE) ALUMINUM-BASED FUEL ELEMENT SCHEMATIC A.5
FIGURE A.4 TYPICAL PIN-TYPE (ALUMINUM-BASED) FUEL ELEMENT SCHEMATIC A.6
FIGURE A.5 TYPICAL INVOLUTE(1)-TYPE (ALUMINUM-BASED) FUEL ELEMENT SCHEMATIC A.7
FIGURE A.6 TYPICAL INVOLUTE(2)-TYPE (ALUMINUM-BASED) FUEL ELEMENT SCHEMATIC A.8
FIGURE A.7 HIGH BURNUP U_3O_8-AL FUEL IRRADIATED IN RESEARCH AND TEST REACTORS A.17
FIGURE A.8 BINARY PHASE DIAGRAM OF THE URANIUM-ALUMINUM SYSTEM A.17
FIGURE A.9 HIGH BURNUP UAL_x-AL FUEL IRRADIATED IN RESEARCH AND TEST REACTORS A.18
FIGURE A.10 HIGH BURNUP U_3Si_2-AL FUEL IRRADIATED IN RESEARCH AND TEST REACTORS A.19
FIGURE A.11 TERNARY ISOTHERMAL SECTION FROM THE U-AL-SI SYSTEM AT 950°C A.19

1.0 EXECUTIVE SUMMARY

The U.S. Department of Energy has selected the Savannah River Site (SRS) as the location to consolidate and store aluminum-based spent nuclear fuel (Al-SNF) from domestic and foreign research reactors. The Research Reactor Spent Nuclear Fuel Task Team, appointed by the Office of Spent Fuel Management of DOE, assessed and identified the most promising technology options for the disposition of aluminum based domestic and foreign research reactor SNF in a geologic repository. The most promising options identified by the task team were direct/ codisposal and melt-dilute technologies. SRS implemented the task team recommendation and initiated an Al-SNF Alternate Technology Program (ATP). The program focuses on the development of the direct/codisposal and melt-dilute technology options for the disposition of the aluminum spent nuclear fuel in a geologic repository.

The Al-SNF ATP consists of both technology development activities and project implementation activities. The technology program activities include development of the treatment process flow sheets and the technical basis for the disposition of Al-SNF in a geologic repository. The current project implementation activities include development of a treatment and storage facility (TSF) to treat, package, and store Al-SNF using the melt-dilute and direct/codisposal technologies.

The primary functions of the SRS TSF is to treat, package, and store Al-SNF in a form ready for disposal in a geologic repository, presently termed the Mined Geologic Disposal System (MGDS). The primary process steps associated with the treatment of Al-SNF in the SRS TSF include:

- receipt from off site,
- temporary basin storage,
- pre-treatment characterization,
- treatment to produce an Al-SNF form,
- packaging the Al-SNF forms into road ready canisters, and interim storage of the canisters at SRS.

Functional performance requirements (FPRs) are being assembled by the SRS Spent Fuel Storage Division to facilitate the design of an Al-SNF Treatment and Storage Facility (TSF) at SRS. The FPRs address each of the process steps for both treatment alternatives, namely melt-dilute and direct/codisposal.

This report presents the technical bases for the FPRs. The technical bases detailed herein demonstrate the viability of the treatment technologies with respect to interim storage requirements, repository disposal requirements and treatment needs. The results from the SRS Al-SNF ATP process development and SNF form qualification studies guide the development of the FPRs.

2.0 INTRODUCTION

2.1 Background

WSRC has been developing the technologies for two alternatives for chemical reprocessing, namely Direct Disposal (DD) and Melt-Dilute (MD), as part of the Alternate Technology Program. Both Al-SNF forms produced using the DD and MD concepts are viable for the ultimate disposal of Al-SNF in the Mined Geologic Disposal System [1].

The SRS Environment Impact Statement (draft)[2] recognizes that one or both of the technologies may be safely used to disposition the Al-SNF from Foreign and Domestic Research Reactors (FRR and DRR, respectively). The DD and MD technology program activities supported the selection of the MD technology as the preferred technology with DD as the backup technology [3]. For the purposes of facility design, a preconceptual design was authorized for the MD option. The baseline design includes a facility in L-Reactor Area that will have a primary MD treatment facility with the option of implementing DD for certain SNF. Functional Performance Requirements have been developed for a TSF to implement the MD technology with DD as a backup [4]. This report presents the technical bases for the FPRs the TSF.

2.2 Evolution of the Alternate Technology Program and Technical Strategy for Disposition of Al-SNF

The United States Department of Energy has selected the Savannah River Site (SRS) as the location to consolidate and store Al-SNF, originating in the United States, from FRR and DRR [5,6,7] through the Environmental Impact Statement (EIS) process. These SNF are either in service, being stored in water basins or in dry storage casks at the reactor sites, or have been transferred to SRS and stored in water basins. A portion of this inventory contains HEU. Since the fuel receipts would continue for several decades beyond projected SRS canyon operations, it is anticipated that it will be necessary to develop disposal technologies that do not rely on reprocessing.

The Research Reactor Spent Nuclear Fuel Task Team was appointed by the Office of Spent Fuel Management of DOE to evaluate the effectiveness, relative merits, costs, and difficulties in implementation of alternative technologies and waste forms for the treatment, packaging, and disposal of aluminum-based SNF [8]. The base case for comparison of the several technologies was chemical reprocessing followed by incorporation into borosilicate glass. The principal recommendations of the Task Team were:

- parallel development of direct disposal and dilution technology options including codisposal with HLW borosilicate glass logs;
- utilization of the SRS canyons for chemical reprocessing of SRS fuel, failed or sectioned fuel, and other selected fuel;
- electrometallurgical treatment as an advanced technology backup to direct and dilution technologies;
- DOE-NRC agreement on requirements for SNF disposal and waste form for HEU; and
- planning, funding, and designing a dry storage facility at SRS.

None of the alternative waste forms other than borosilicate glass resulting from processing are qualified for disposal in the federal repository in Yucca Mountain.

Development and licensing of the facilities of a Mined Geologic Disposal System or the repository for ultimate disposal of SNF and high level waste (HLW) is the responsibility of the Office of Civilian Radioactive Waste Management of DOE. At present, a licensing application for the disposition of SNF and high level waste glass in the proposed repository at Yucca Mountain is expected to be completed by 2002. The licensing application drives the schedule for the repository Performance Assessment and the Viability Assessment to be completed during the 1998 - 2002 period. In order to ensure that Al-SNF disposition is part of the DOE-RW licensing application, a technology development program aimed at evaluating non-processing alternatives was initiated in FY97. The purpose of this program is to complete the engineering analyses to support the FPR development and develop the scientific bases necessary to ensure qualification of an appropriate Al-SNF form for the repository. The two candidate forms are the direct disposal form (i.e., Al-SNF assemblies) and the melt-dilute form. The direct disposal form involves the placement of Al-SNF in baskets within a DOE SNF canister. The melt-dilute form involves treatment of the Al-SNF to create a monolithic product of a combination of Al-SNF, depleted uranium, and aluminum.

2.3 Report Organization

This report will provide the technical bases for FPRs for the processing of Al-SNF for ultimate disposal using both the melt-dilute and the direct disposal concepts. The FPRs that are addressed in this report are those derived from interim dry storage, transportation, or repository requirements, or that are needed for the treatment of Al-SNF. The complete set of FPRs is contained in Reference 4.

Chapter 3.0 contains a cross-reference table that relates the FPRs in Reference 4 to the sections within this report that develop the technical bases. Chapter 4.0 of this report describes the process steps that use the melt-dilute technology for treatment and storage of Al-SNF in canisters, ready for repository disposal. Chapter 5.0 describes the process steps that use the direct disposal technology for treatment and storage of Al-SNF. Appendix A covers the scope of Al-SNF materials to be received and handled in the TSF at SRS.

Numerical values in statements in the report labeled "TBV" are those labeled as such in the requirements for the MGDS [9]. In addition, it must be noted that requirements from which the FPRs are developed are subject to change.

2.4 References

1 WSRC-TR-98-00227, Disposability Assessment: Aluminum-Based Spent Nuclear Fuel Forms, June 1998.

2 SRS Environmental Impact Statement, draft

3 U-ESR-G-00004, "Spent Nuclear Fuel Alternate Technology Decision Analysis," E. W. Zimmerman, June 26, 1998.

4 WSRC-TR-98-00218. Functional Performance Requirements for a Proposed Treatment and Storage Facility for Spent Nuclear Fuel, July 1998.

5 *Final Environmental Impact Statement, Interim Management of Nuclear Materials*. DOE (U.S. Department of Energy), DOE-EIS-0220, Savannah River Operations Office, Aiken, South Carolina (1995).

6 Final Environmental Impact Statement, Proposed Nuclear Weapons Non-Proliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel. DOE (U.S. Department of Energy), DOE/EIS-0218F, Assistant Secretary for Environmental Management, Washington, DC (1995).

7 *DOE-Owned Spent Nuclear Fuel Technology Integration Plan*. DOE/SNF/PP-002, Rev. 1, Idaho National Engineering Laboratory (May 1996).

8 "Technical Strategy for the Treatment, Packaging, and Disposal of Aluminum-Based Spent Nuclear Fuel", Volume 1 of the *Research Reactor Spent Nuclear Fuel Task Team Report*. prepared for the Department of Energy - Office of Spent Fuel Management (May 1996).

9 M. S. Abashian. "Mined Geologic Disposal System Draft Disposability Interface Specification," DI Number B00000000-01717-4600-00108 REV 00. Civilian Radioactive Waste Management System Management & Operating Contractor (February 1998).

3.0 CROSS-REFERENCE TO FPR

Table 3.1 Cross-Reference between Functional Performance Requirements Document and Technical Bases Document (TBD)

FPR Description	TBD Section
Conditioning SNF For Melt-Dilute Treatment	
Dry SNF before loading into furnace	
Heat SNF to 200°C for 2-4 hours to remove free water	4.2
Provide confinement, filtering, and monitoring of drying offgas exhaust	4.2
Weigh SNF	
Weigh SNF weighing up to 13 kg with an accuracy of ± 1-kg	5.2
Conditioning SNF For Direct Codisposal	
Dry SNF for Direct Codisposal	
Vacuum drying requirements To Be Determined	5.4.1-5.4.3
Provide confinement, filtering, and monitoring of drying offgas exhaust	5.4.1
Weigh SNF	
Weigh SNF weighing up to 13 kg with an accuracy of ± 1-kg	A.4.1-A.4.3
Melting and Isotopic Dilution	
Melt and alloy the SNF listed Appendix F	
The melting furnace shall be operated in a batch mode. The maximum batch size shall be 25 assemblies	4.4.4.3
The melting furnace shall be capable of processing one batch per operating shift	4.4.4.3
Mixing SNF types within a batch shall satisfy the following requirements: No Mixing	4.4
The melting furnace shall be capable of stirring the molten constituents to produce a homogeneous mixture defined as To Be Determined	4.6.2
The melting furnace shall be capable of melting and heating 300-400 kg of aluminum from ambient temperature to 800-1000°C in 2-3 minutes	4.4.4 and 4.6
The melting furnace shall provide the capability of measuring the molten alloy temperature up to 800-1000°C with an accuracy of ± 50°C	4.4 & 4.4.4
Determine the composition of molten SNF before and after addition of aluminum and depleted uranium feed materials	
Determine the U-Al composition, accurate to within ± 5 wt% U-Al	4.4.1
Determine the ^{235}U enrichment (^{235}U weight fraction of total uranium) accurate to within ± 5 wt% ^{235}U	4.1 & 4.4.1
Determine the concentrations of impurities accurate to within ± 5 wt% for the sample taken after the addition of aluminum and depleted uranium feed materials	4.5
Return alloy sample residues to the furnace for melting with SNF in subsequent batch(s)	4.5
Solidify the molten alloy	
Provide sufficient convection cooling to solidify the molten alloy within two hours	4.4.3
Identify solidified waste form.	
Identify each solidified waste form with a permanent, unique identification number	4.9.4

FPR Description	TBD Section
Weigh solidified SNF Form	
Weigh solidified SNF forms weighing up to 150 kg with an accuracy of 10 kg	4.4.4.3, A.4.1, A.4.2, & A.4.3
Characterization	
Verify SNF identification number	
Remotely read and verify SNF identification number prior to conditioning	5.2
Remotely read and verify SNF identification number prior to Melt-Dilute	5.2
Remotely read and verify SNF assembly or SNF form identification number prior to placing into canister	5.2
Characterize SNF	
Determine radionuclide inventory	4.5 & 5.3.2
Determine SNF heat output	4.9.3 & 5.6.3
Determine SNF (Direct Codisposal only) radiation intensity (γ, neutron)	5.6.3
Packaging	
Package SNF For Direct Codisposal	
Baskets, spacers, neutron absorbers	5.6.1
Remove hydrides from SNF	
Vacuum dry to $\leq$ 5 torr for > 15 minutes at > 25°C	5.4.3
Seal the canisters for storage and transportation	
Weld	4.9.2 & 5.6.2
Evacuation and Inert Gas Backfill	4.9.2, 5.6.2, & 4.10.1
Leak Testing	4.9.2 & 5.6.2
Determine weight of canister	
Determine weight of sealed canister, weighing up to 2250 kg with an accuracy of ± 50 kg	4.9.3 & 5.6.3
Determine radiation intensity of canister	
Determine radiation intensity (γ, neutron)	4.9.3 & 5.6.3
Open sealed canisters and remove SNF	
Provide capability to open sealed canisters and remove SNF	4.10.1
Road Ready Storage	
Provide interim storage of sealed SNF canisters until the geologic repository is ready to receive SNF	
The Road Ready Storage shall be capable of storing up to 600 sealed SNF canisters	4.1
The Road Ready Storage shall be capable of retrieving stored canisters from the storage modules and returning them to Building 105-L	4.10.1
Convection cooling	
Provide natural convection cooling to maintain SNF (or canister) temperature < 200°C	4.10.1

4.0 MELT-DILUTE PROCESS STEPS AND FPRs

The melt-dilute treatment technology is the preferred alternative treatment technology [1]. The overall process is subject to federal and site requirements and treatment needs to which FPRs are developed. The process steps are show in Figure 4.1 below.

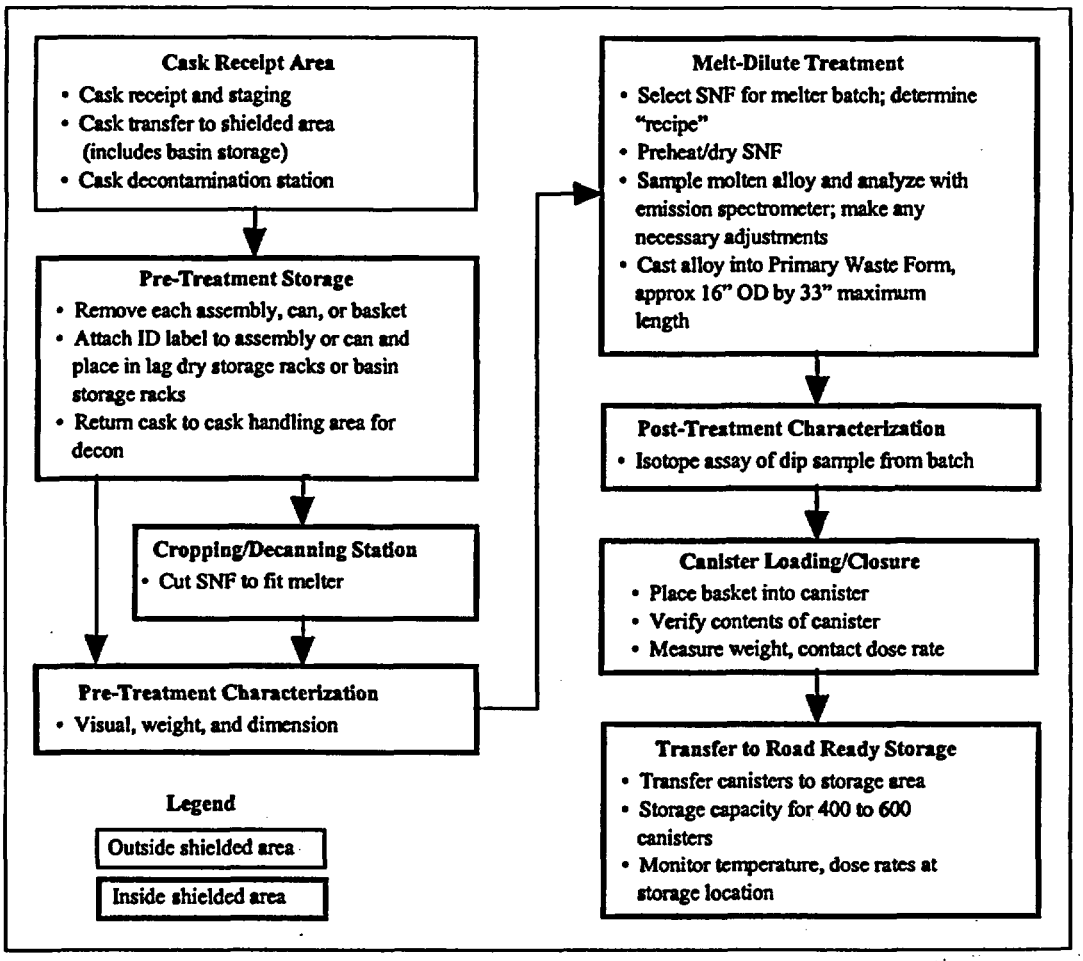


Figure 4.1 Process Steps in TSF for Melt-Dilute Technology

4.1 Melt-Dilute Treatment Overview

Many MTR and other research reactor fuel elements were originally fabricated using 93% highly enriched uranium and aluminum alloy, and they were generally clad with 6061 aluminum. These fuel assemblies have been irradiated, and some have been stored for up to 40 years in water basins after being discharged from the reactors. During irradiation, approximately 30-60% of the original ^{235}U content of the fuel assembly is consumed. To dilute the enrichment and reduce criticality and proliferation concerns for long-term repository storage, the melt-dilute technology was proposed for treatment of spent nuclear fuel. When SNF assemblies are melted, the composition of the molten alloy is expected to range from 3-10 wt% uranium in aluminum. For dilution, depleted uranium metal is added to the melt to reduce the enrichment of the uranium to about 20% or less. Aluminum can be added to the melt, if needed, to adjust the casting

composition of the alloy in the crucible to process requirements, such as the eutectic or the intermetallic composition.

When depleted uranium metal is added to dilute the fissile content to ≤ 20%, the phases present in the melt and in the final microstructure of the casting are affected. According to the phase diagram in Figure 4.2, phases vary from liquid to liquid plus solid as the uranium content increases at temperatures above 660°C. Once inside a two-phase region, the solid intermetallic phases of either primary UAl_3 or UAl_4 become thermodynamically stable and begin to precipitate from the liquid solution. Because the density of the intermetallic is greater than the liquid alloy, they tend to segregate at the bottom of the crucible unless stirred continuously. Heating the melt above the liquidus temperature dissolves the intermetallic compounds and forms a single liquid phase. Above the liquidus temperature, there is complete isotopic homogenization or uniform dilution of the uranium fuel.

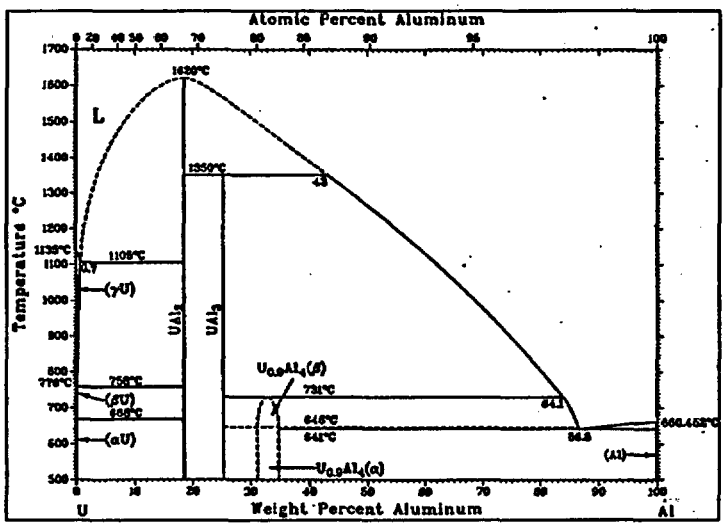


Figure 4.2 Binary Phase Diagram of the Uranium-Aluminum System

It should be noted from Figure 4.2 that when the composition of the alloy exceeds 30 wt% uranium, the liquidus temperature becomes greater than 1000°C and can reach 1500°C for a 67 wt% alloy of the intermetallic UAl_4 composition. At these temperatures, the release of fission products from the melt increases as will be discussed in a later section. The solubility of hydrogen in the melt also increases with temperature causing an increased potential for porosity in the solidified structure.

There are various uranium compositions that can be cast for the uranium-aluminum system, but for the melt-dilute process, a specific alloy content or an adjusted composition is desired for the waste form. This simplifies both the production process and waste form qualification for the geologic repository. The composition selected for the process basically depends on the isotopic dilution required for the spent nuclear fuel waste form and the waste volume criteria for repository storage. Three basic isotopic dilutions have been investigated and include 20%, 5%, and 2%. Each dilution option can be achieved by either adding depleted uranium metal or by adding both depleted uranium and aluminum metal to the molten alloy. The process used for treatment affects the waste volume as indicated in Figure 4.3 where the waste volume is shown as a function of the dilution factor (%) for aluminum-base SNF. Advantages of diluting the uranium

to below 20% ^{235}U and the eutectic composition include: (1) lower process operating temperatures, (2) minimum gravity segregation in the casting, (3) lower volume of offgas products and (4) lower associated costs. When compared to other dilution methods, the 20% dilution offers the greatest versatility because waste forms containing approximately 5-67 wt% uranium can be produced and stored in less than 400 canisters each. If the waste form is diluted to approximately 20% ^{235}U, criticality and proliferation concerns are reduced or eliminated for geologic repository storage.

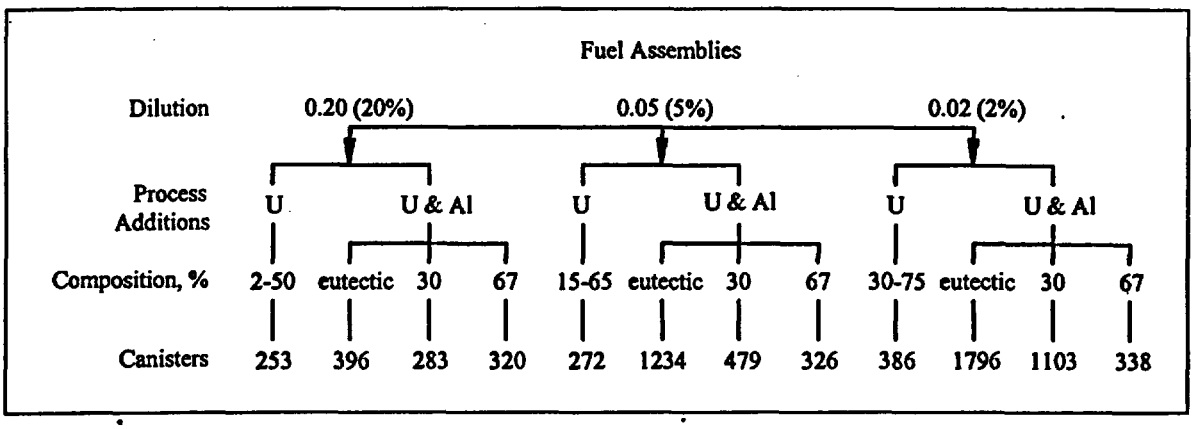


Figure 4.3 Total Waste Volume for FRR, DRR, HFIR, and ATR Fuel Assemblies as a Function of the Dilution Factor, Process Additions and Waste Composition [2].

4.2 Pre-Treatment Drying

Because the melt-dilute process requires that the fuel be melted, it is imperative that it be dry. The introduction of liquid water into molten metal can result in a steam explosion.

4.2.1 Background

A majority of the Al-Based SNF to be received from both domestic research reactors and foreign research reactors has been wet stored in basins with varying degrees of basin water chemistry control. Additionally, the current plan for the TSF involves removing fuel from the L-Basin directly to the TSF within the 105-L reactor building. Thus, the initial step in the treatment of the SNF must involve a drying process.

The drying procedure for the melt-dilute process is simple. The process involves a basic pre-treatment drying oven that will be used to dry the fuel at a temperature in the range of 150-200°C. The recommended drying scenario can be applied to all the fuel types regardless of fuel core alloy or design. The oven will require the appropriate offgas system for the gases expected at up to 200°C, which is primarily krypton.

4.2.2 Preventing Melt-Water Interactions

For the MD process, it is necessary to dry the SNF assemblies prior to treatment to preclude the possibility of melt-water interactions. Melt-water interactions or steam explosions have the potential to effect injuries, fatalities, or loss of equipment. As metal containing surface water or water trapped in crevices, etc. is added to molten alloys, the free water instantaneously vaporizes

with a subsequent volume expansion resulting in a steam explosion. With this steam explosion, molten metal is expelled rapidly from the crucible. Work at Oak Ridge National Laboratory during the 1960's showed that the violent nature of steam explosions increases with increasing melt temperature and water volume [3,4,5].

The drying stage within the MD process seeks to prevent or eliminate the possibility for a steam explosion by removing the free water from the SNF assemblies prior to contact with molten metal. Fuel will be removed from the basin and allowed to "drip" and air-dry in Lag storage prior to movement of the assemblies to the treatment room of the TSF. This will result in only minimal free surface water for removal with only crevice or trapped water to be removed via a drying operation.

4.2.3 Pre-Treatment Furnace Drying

Common practice within the commercial aluminum foundry industry is to use a drying oven to remove moisture from process crucibles, charge materials, and casting utensils (dip ladles, slag rakes, and thermocouples). These ovens are typically temperature limited (< 400°C) and resistance heated. Some models use a heated forced air convection arrangement to remove moisture.

There are many different sizes of fuel to be processed, but all fuel will eventually be melted in the same furnace. The largest fuel assemblies such as HFIR and RHF will be processed one at a time, because only one will fit into the furnace crucible. Smaller assemblies will be bundled into groups large enough to just fit into the crucible. Before a fuel element, or bundle of elements, is moved to the melt furnace crucible, it must first be dried in a drying oven.

The drying oven should be designed to be large enough to hold the bundles of fuel that will be processed during one day. It must be designed such that it is inherently critically safe under all operating conditions. The oven will operate at a slight negative pressure with the offgas being vented to an exhaust system containing a HEPA filter that is designed to accept moist air that is possibly contaminated with fission gases, tritium, and/or particulate material.

The temperature of the oven containing the fuel or fuel bundle will be increased to 200°C and held at that temperature for a time, ~2 hours, in excess of that demonstrated to remove all liquid water. Fuel will be moved directly from the drying oven to the melt-dilute furnace.

4.3 Pre-Treatment Characterization

One of the major challenges to the development and final selection of an approved alternate technology for the disposition of DOE-owned Al-clad spent nuclear fuels is the lack of clear definition of the requirements for repository acceptance of the fuels. Although the National SNF Program is currently developing guidelines for SNF custodians to follow in the preparation of fuels for repository acceptance, these guidelines are preliminary in nature and are currently lacking adequate detail as to the appropriate test techniques, methodologies, repository conditions, and acceptable accuracy/precision levels for form assessment. This challenge is amplified by the fact that the fuel property data and operation history of the fuels to be provided by SNF custodians may also be inaccurate, lacking in detail, or in some extreme cases, non-existent. Characterization for these fuels is therefore anticipated to be required to a greater extent, if only for documentation purposes.

Although the validity of data (Appendix A-type) provided by SNF custodians is currently unknown or is at least considered subjective, this issue may be negotiated between the OCRWM,

NRC, and fuel custodians. Until a definitive official decision or agreement is made by the respective responsible agencies, however, the pretreatment characterization requirements for either the direct disposal or melt-dilution disposition options cannot be clearly defined and are heavily dependent upon the nature of such an agreement.

4.3.1 Melt-Dilution

The primary advantage of the melt-dilution process, however, is that if the target composition can be shown through experimental, analytical, or computational means to consistently exhibit characteristics far below prescribed limits, the need for pre-treatment characterization on a per - melt basis will be reduced. It should be made clear, however, that although the melt-dilution process essentially "erases" the fuel assembly history and thereby creates a new form, the final waste package must meet the repository acceptance criteria. Post-treatment characterization of the form may have to be performed to establish that the waste package meets such criteria as thermal load, radiation level, IAEA safeguards/security traceability and/or accountability requirements, fissile species, etc.

4.3.1.1 Background

The SNF form is significantly altered in the MD process and is considered to be inherently independent of the fuel fabrication and operation history. Hence, the number of pre-treatment characterization activities to be performed is significantly reduced.

For the melt-dilution process, the key parameters for process control are the total uranium content and the enrichment, which are used as input for melt composition calculations to determine the relative amounts of depleted uranium, aluminum, and/or poisons to be added in order to achieve the final target composition. These parameters may be determined from review of the fuel data package (Appendix A-type) or may be determined through analytical measurement in the melt during processing. Although acceptance of Appendix A data would be beneficial, determination of these parameters during actual processing may be preferred in order to obtain a more accurate value for the homogeneous form. Therefore, the melt-dilution process essentially eliminates the need to know the in-depth history of the fuel but does necessitate an in-melt characterization.

4.3.1.2 In-Process Characterization Requirements

The melt-dilution option results in a completely different form, which may exhibit superior properties in terms of corrosion resistance, leachability, etc. SNF assemblies (MTR-type) are to be placed into a furnace (final design to be determined), melted and mixed with depleted uranium and/or aluminum and/or poison to alter the final composition, and then cast into a mold or crucible. A eutectic (13.2 wt%) composition has been selected and the program is currently focusing on the selection of a 20% (or less) ^{235}U composition to reduce criticality concerns associated with higher enriched fuels. It is expected that the enrichment will be between 10% and 20%; the exact enrichment will be chosen on the basis of criticality analysis and a costs benefits analysis. Depending upon the SNF form constitution, melt temperatures, furnace design, crucible material selection, and offgas systems will be affected. Assemblies are assumed to be melted and processed one at a time, filling approximately 80% of the canister volume.

From the total U and enrichment data, a source term using ORIGEN-S or similar validated computer code may be employed to produce the isotopic inventory of the pre-treated fuel. Upon melting however, some isotopes and reactive species may be driven off during the melting and mixing process, thus altering the isotopic inventory of the final product. If, however, the target form composition is less than or equal to 20% ^{235}U, the form may not necessarily have to conform to the stricter standards that apply to HEU fuel materials (> 20% ^{235}U) such as material

accountability. While the probability of criticality is greatly reduced, some degree of criticality controls may remain necessary.

The current strategy for process control is to measure total uranium concentration as well as uranium isotopic (238 and 235) concentrations. Dip samples taken form the melt will be utilized for these measurements. The technique used to perform these measurements should have a quick response time in order to avoid long holding periods during the treatment process operation. Current studies show that it is possible to measure directly the total uranium and isotopic concentrations using a spark arc, glow discharge, or laser ablation emission source coupled with inductively coupled plasma emission spectroscopy. This system can provide fast response with accuracy in the range of 5-10%.

4.4 Al-SNF Treatment and Product Consistency

This section provides details concerning pertinent characteristics of the spent fuel inventory and the uranium-aluminum alloy system. Additionally, potential process/casting options are discussed as well as the overall benefits of this treatment technology..

4.4.1 Candidate Uranium Alloy Compositions

There are currently three fuel types that must be treated using the melt-dilute treatment process. Of these, the binary U-Al_x fuels comprise the largest quantity both foreign and domestic. The uranium silicide and oxide fuels account for roughly 25% of the total inventory combined. The first step in the development of the melt-dilute treatment process was to determine an appropriate alloy composition(s) since many processing variables such as temperature and time are highly dependent on composition. The following section will discuss the candidate alloy composition for the three fuel types.

4.4.1.1 U-Al SNF

Initially, three alloy compositions were considered for the melt-dilute treatment of binary uranium-aluminum alloy fuels. They are the eutectic composition, 30 wt% uranium-aluminum, and the intermetallic composition of UAl_4. The eutectic composition is of particular interest because it has the lowest liquidus temperature in the U-Al alloy system. The 30 wt% uranium alloy has a liquidus temperature of less than 1000°C, and UAl_4 has the lowest liquidus temperature of the intermetallic compounds.

The eutectic composition contains about 87 wt% aluminum and only 13 wt% uranium. Aluminum must be added to the melt for diluted alloy compositions above the eutectic. The amount of aluminum depends on dilution requirements of the process and tends to increase the waste volume. Aluminum adds about 370 cc/Kg to the waste volume, whereas uranium adds only about 50 cc/Kg. However, solidification of the eutectic composition is unique in that it occurs at a constant temperature of 646°C. The solidification characteristic of the eutectic produces a uniform microstructure with little or no gravity segregation of the aluminide intermetallic phase. Because the composition range is narrow, the microstructure generally consists of a small amount of primary aluminum or UAl_4.

For the 30 wt% uranium composition, the liquidus temperature is about 1000°C, and there is a 350°C two-phase zone where separation of primary aluminides can occur during solidification. The aluminides are dense and will separate from the liquid without stirring and settle to the bottom of the casting during slow cooling. For alloys melted in graphite crucible and poured into graphite mold, it has been shown that as much as a 5% difference in primary aluminide concentration can exist for a 30 wt% U alloy.

The uranium-aluminum alloy system has three different intermetallic compounds, UAl_2, UAl_3, and UAl_4. The composition of UAl_4 is about 67 wt% uranium and 33 wt% aluminum. The liquidus temperature for this composition is approximately 1500°C, and the peritectic reaction temperature is 731°C. The low temperature of the peritectic reaction requires a relatively long diffusion time at temperature for completion. Thus, the microstructure of a slow cooled casting most often consists of a mixture of both UAl_3 and UAl_4 intermetallic compounds. These compounds generally have different physical and chemical characteristics that differ from the solid solution, which could prove beneficial for the corrosion resistance of the waste form.

The current recommended melt-dilute treatment process alloy composition is the eutectic with dilution to ≤ 20% ^{235}U. As previously mentioned the eutectic composition of 13.2 wt% U has the lowest liquidus temperature on the U-Al Phase diagram. This will enhance the remote operability and safety of the treatment process as well as limit the radionuclide species, which will volatilize from the melt.

4.4.1.2 U-Al-Si SNF

The United States Reduced Enrichment Research and Test Reactor program operated by Argonne National Laboratory has worked for the past 15 years to develop low enriched (< 20%) uranium alloy fuel for use in research and test reactors. The main driver for this program is the development of non-proliferable nuclear fuels. In order to lower the ^{235}U enrichment, the total uranium concentration within the fuel meat must be increased. Increasing the total uranium concentration within the fuel core results in difficulties of manufacturing due to loss of ductility with increasing uranium loading. Conventional fuel manufacturing techniques such as rolling and extrusion are not possible with the current fuel alloys at such high uranium loading. Thus, the RERTR program sponsored several research initiatives to investigate new fuel forms that can be manufactured with current industry techniques. The alloys system that met with the greatest success is the U-Si binary system. Examination of the U-Si binary equilibrium phase diagram shows two compounds possessing high atomic uranium fraction: U_3Si and U_3Si_2. Research and test reactor fuels using either of these two compounds are manufactured as fuel core composites consisting of powder metallurgy blends of Al powder and the individual uranium-silicide compounds. These powder composites are generally roll-bonded with Al-cladding alloy. Currently, U-Al-Si alloy fuels are used in certain reactors and are expected to comprise approximately 15% of the fuel inventory to be treated by the melt-dilute process. A photomicrograph of the fuel core from a typical silicide fuel assembly showing interaction between the uranium-silicide intermetallic and the Al matrix is shown in Figure 4.4. The new phase formed in the reaction zone is the $U(Al,Si)_3$ which is one of the equilibrium phases for this composition.

Calculation based on aluminide fuels for the silicide loading indicates that melting of a U-Al-Si fuel assembly will result in a melt-composition in the range of approximately 1-10 wt% uranium, 1 wt% Si and 89-98 wt% Al for the various MTR assemblies. For the melt-dilute process alloy composition is the main driver for treatment temperatures. For a ternary U-Al-Si in the indicated concentration range, little if any ternary liquidus information was available. The only published data available indicates a liquidus temperature between 660°C and 950°C for an alloy with this composition. Ternary isothermal sections were developed for the U-Al-Si system using the Gibbs triangulation method and experimental data in order to determine a more accurate melting and liquidus information. Ternary isothermal sections from 500-800°C are depicted in Figure 4.5 and show the relative position of the expected fuel composition within the phase fields as temperature

increases. From these isothermal sections it can be seen that a typical 90Al-9U-1Si alloy will be fully molten at 800°C.

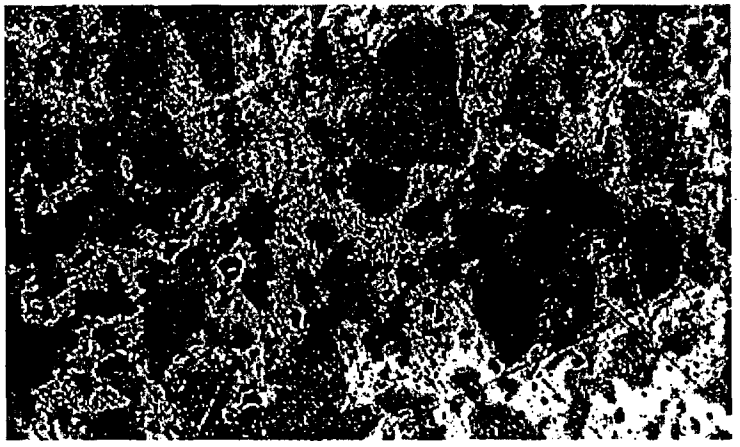


Figure 4.4 Post-Irradiation Microstructure of a U-Al-Si LEU Fuel

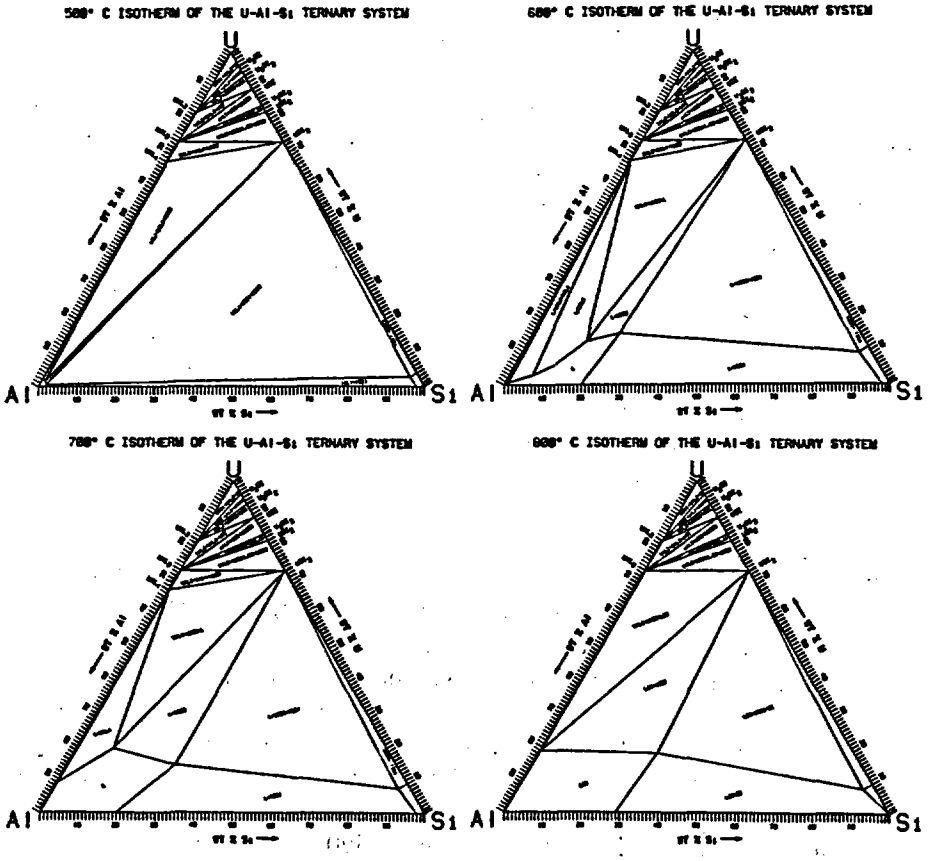


Figure 4.5 Ternary Isothermal Sections for the U-Al-Si Alloy System

4.4.1.3 U_3O_8-Al SNF

Much like the silicide fuels, the initial development of the uranium-oxide fuel stemmed from a desire to lower research and test reactor fuel enrichment to produce non-proliferable fuels. Thus, the first assemblies produced containing U_3O_8-Al cores were either low enriched (20% ^{235}U) or medium enriched (≅ 45% ^{235}U). Later, however, highly enriched U_3O_8-Al cores were made and constitute the bulk of the oxide assemblies in service. Like the silicide fuels the uranium oxide fuel particles react with the aluminum matrix while in service as provided in Figure 4.6. However, the extent of the reaction in the oxide is much greater than in the silicide fuels. Identification of the phases present in the reaction zone has been difficult and limited. UAl_x compounds and aluminum oxide phases have been reported to be present in the reaction zone.

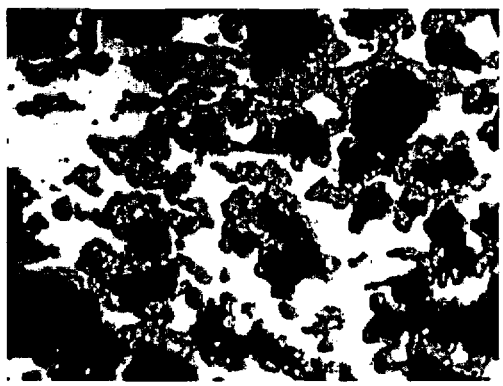


Figure 4.6 Post Irradiation Microstructure for a U_3O_8-Al HEU Fuel

For the operation of the melt-dilute treatment process, the composition of a typical oxide fuel assembly, with melting and full reaction of the oxide particles with the aluminum has been determined using a HFIR core as representative. The final alloy composition for a fully molten and reacted HFIR core is approximately 90-93 wt% aluminum, 6-10 wt% uranium and 1-2 wt% oxygen. In keeping with the binary UAl_x fuels the expected final melt-dilute treatment process ingot composition will contain ~13 wt% uranium. By keeping the uranium concentration the same for the two fuel types the final microstructures and performance properties will remain relatively similar.

4.4.1.4 Sterling Forest Oxide Type SNF

Sterling Forest oxide type powder consists of a mixture of U_3O_8 and UO_2 oxides. These materials are currently stored in aluminum can and can be readily treated using the melt-dilute treatment process. The treatment process steps and parameters will be similar to the oxide fuel. The final diluted product will, like the oxide fuels, have a composition of approximately ~13 wt% U for consistency.

4.4.2 Process Cycle Options

In general, casting processes or cycles commonly found in commercial foundries employ either superheat casting or mushy-zone casting. Superheat casting involves heating the desired alloy to some prescribed temperature above the alloy liquidus temperature (superheating) to ensure maximum homogeneity and distribution of the alloying elements. Mushy –zone casting, on the other hand, involves heating the alloy to a temperature below the liquidus within a two-phase region where one of the two phases is a liquid and the other is a solid. Two potential drawbacks

to this latter casting cycle are the need for longer alloying times and the relative inability to ensure alloy homogeneity. While both of these techniques have been explored for the melt-dilute treatment process, a casting process cycle, which involves a fully homogenous molten liquid, has been determined to be optimal and as such the superheat casting technique is recommended. Additionally, an upper bound to the treatment cycle temperature of 1000°C has been established in order to ensure ease of remote process operation and safety and also to reduce the necessary complexity of the offgas system. The following sections will discuss the schematic treatment cycle as well as any cycle adjustment/adaptation necessary for each fuel type.

4.4.2.1 U-Al SNF

For the melt-dilute treatment process, a schematic of the treatment cycle is shown in Figure 4.7. The basic schematic shows heating of the alloys above the liquid temperature but not to exceed 1000°C. Once at temperature it is envisioned that depleted uranium and aluminum will added to achieve the desired final alloy composition and process sampling will be performed to monitor and control the treatment process.

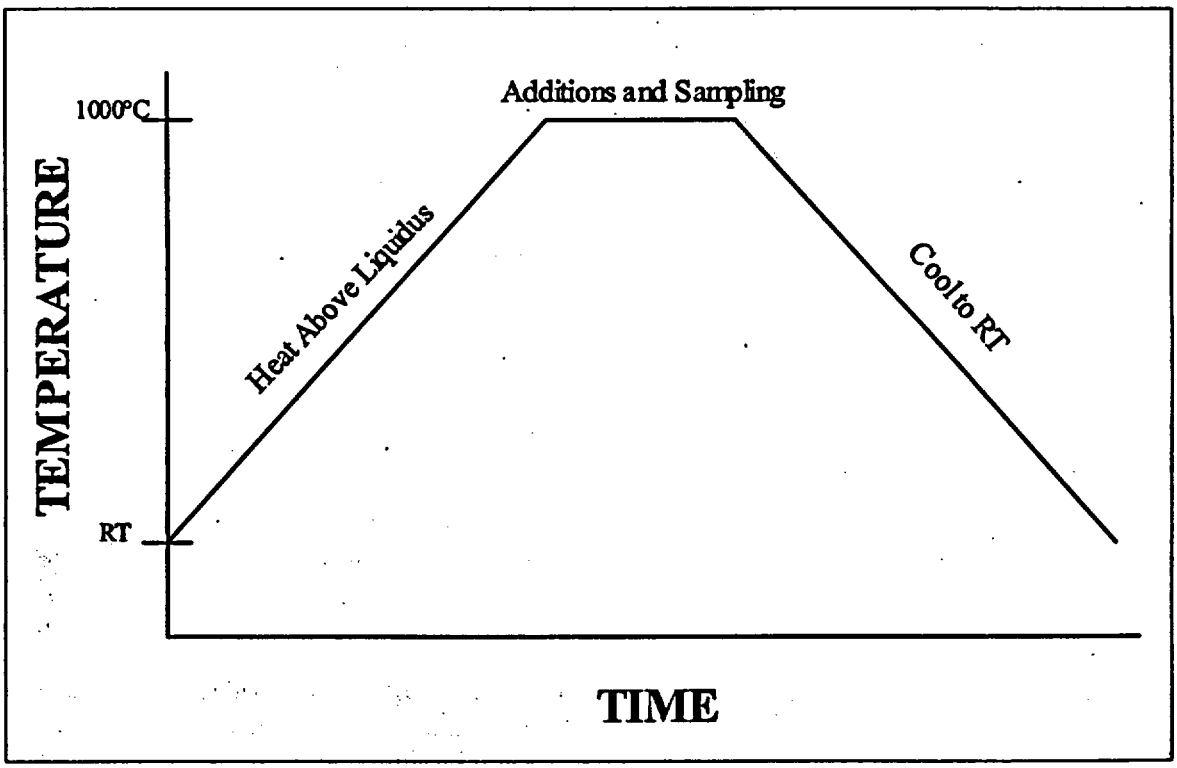


Figure 4.7 Schematic Treatment Cycle for the Melt-Dilute Treatment Process

For the binary U-Al fuels the desired final alloy composition as previously specified is the eutectic composition. The liquidus temperature for a eutectic alloy is 646°C. Thus, the treatment cycle temperature will be in the range of 800-850°C to allow for proper melt homogenization. Once the highly enriched assemblies are melted, depleted uranium for dilution and aluminum for composition control are added to the melt. Sampling of the melt both prior to and following the additions of the uranium and aluminum will be performed for process control and verification. The remaining sampled material will be used for post-treatment characterization. The casting

will then be cooled to room temperature and then be removed from the crucible and transported to the packaging station. Processing time is expected to be less than 2.5 hours.

4.4.2.2 U-Al-Si SNF

Based on the discussion in Section 4.4.1.1, it was decided that the ideal treatment temperature range would be less than 1000°C. From the ternary isothermal section developed to determine the liquidus temperature in the previous section, it is obvious that no treatment cycle temperature adaptations are necessary for U-Al-Si fuels.

A further consideration with the treatment of the silicide is the kinetics of dissolution of the uranium silicide intermetallic particles (USi_3 and U_3Si_2) in molten aluminum at temperatures below 1000°C. Dissolution of the intermetallic particles will occur by the following reaction:

$$U_3Si_2 + Al \rightarrow U(Si, Al)_3 + Al$$

Experiments were conducted to evaluate the forward reaction kinetics of this reaction in the desired treatment cycle temperature range. Depleted U_3Si_2 powder was added to molten aluminum at 850°C. Figure 4.8 displays the metallographic results from the sample heated at 850°C for 30 minutes. The results indicate that the U_3Si_2 powder has fully reacted with the molten aluminum. Thus the reaction kinetics will not limit the treatment cycle.

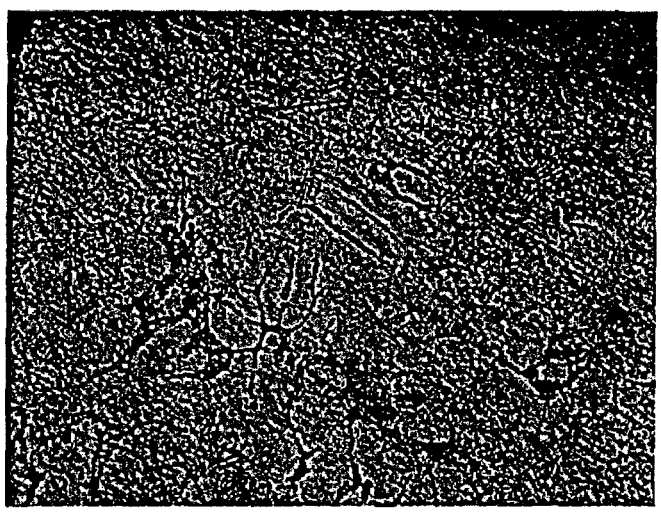


Figure 4.8 Fully Reacted U_3Si_2 Powder in Molten Al at 850°C in 30 minutes

Some question has arisen as to the need for melting and diluting silicide fuels since the as-manufactured enrichment is already below 20% ^{235}U. Melting of these fuels does, however, provide the advantage of volume consolidation which will reduce the total number of canisters needed for geologic disposal compared to the direct disposal option. Additionally, preliminary criticality calculations for the case of a non-degraded repository waste canister indicate the potential to alleviate all concern for a criticality event at enrichment levels between 10-15% ^{235}U. Thus, these silicide LEU fuels could be diluted with depleted uranium to adjust the enrichment to these levels. The impact of a dilution operation on these fuel will not impact the composition or treatment cycle temperature limits for these fuels since the overall composition 90Al-9U-1Si will be maintained with the appropriate alloying additions.

4.4.2.3 U_3O_8-Al SNF

The development of a treatment cycle for the oxide fuels within the bounds of an upper treatment cycle temperature of 1000°C is currently underway. The necessary treatment temperature for both the UAl_x and U-Al-Si fuels has been shown to be in the range of 800-850°C. Although an oxide fuel treatment cycle at a similar temperature is desired, a trade off between treatment cycle temperature and treatment cycle time is necessary to ensure a cost-effective process.

Recent experiments to develop an understanding of the time-temperature relationship for the dissolution of U_3O_8 particles in molten aluminum have been performed. Specimens of depleted U_3O_8-Al fuel tube were heated for times from 1-4 hours in molten aluminum baths. A determination of the driving force for the dissolution reaction based on fundamental thermodynamics has shown that for the oxide fuels the expected reaction is as follows:

$$Al + U_3O_8 \rightarrow UAl_4 + Al_2O_3 + Al$$

This reaction should proceed for temperatures from 800-1000°C. However, thermodynamics provides no indication of the reaction kinetics. Experimental data along with theoretical reduction calculated using the Johnson-Mehl-Avrami model [6] for kinetic behavior are plotted in Figure 4.9. Seventy-five percent of the oxide particles have reacted at 850-1100°C in 3-1 hours, respectively. For the treatment cycle temperature upper bound of 1000°C the model predicts 90% reaction in approximately 3 hours. It appears that treatment of the oxide fuels to effect complete dissolution of the uranium oxide particles and allowing isotopic dilution of the enriched uranium can be achieved without any treatment cycle temperature adjustments.

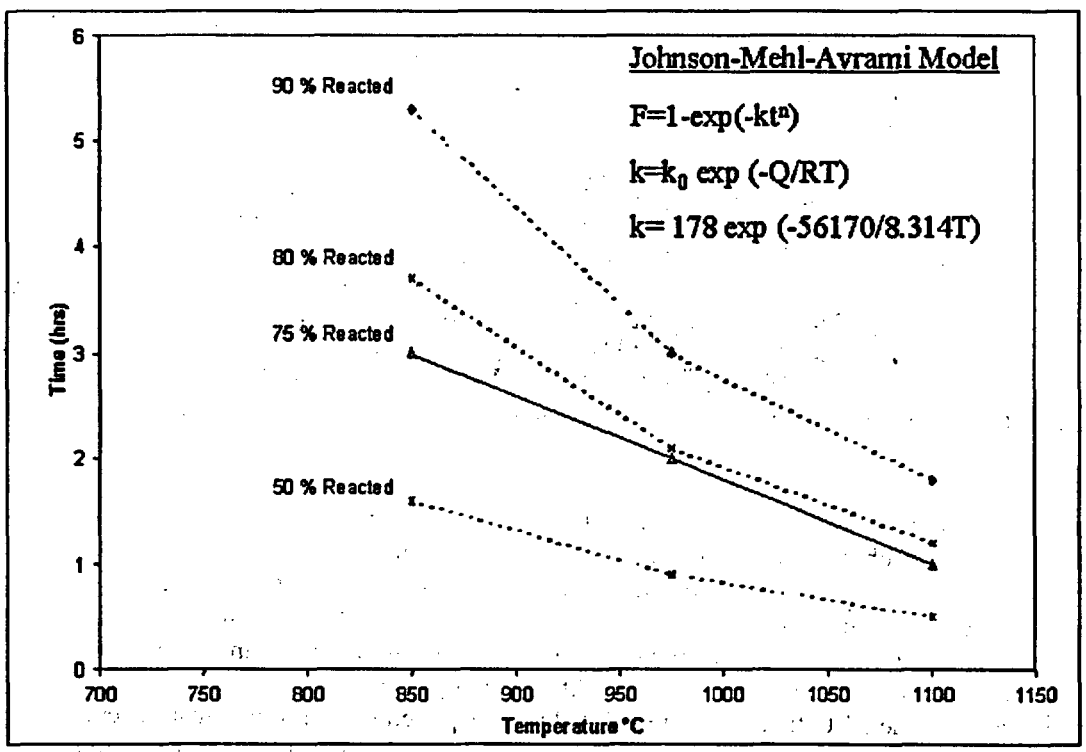


Figure 4.9 Experimental Data and Johnson-Mehl-Avrami Model Predictions of Oxide Dissolution in Molten Aluminum

4.4.2.4 Sterling Forest Oxide-Type Material

Since the sterling forest type oxide powder is comprised of a mixture of U_3O_8 and UO_2 particles, the treatment cycle temperatures and times will be similar to the previously discussed oxide fuels. Like the oxide fuels, the ability to dilute these highly enriched powders isotopically will depend on the kinetics of the oxide dissolution in molten aluminum. One difference with these materials is the presence of the UO_2 particles that will also need dissolution. Calculation based on fundamental thermodynamics shows that the following reaction occurs:

$$Al + UO_2 \rightarrow UAl_4 + Al_2O_3 + Al$$

These oxides are presently stored in aluminum canisters. The oxide powders and aluminum canisters can be added directly to the melt without decanning the powders. Venting the canister, however, is recommended to avoid any possibility for pressurization during heating, resulting in an unsafe condition. Hence, the primary consideration that arises in treating the Sterling Forest oxides is the need to puncture or vent the can in which the material is shipped prior to drying.

4.4.3 Melt-Dilute Casting Techniques

Production of the melt-dilute alloy may be done by one of three casting methods, two of which involve a separate crucible and solidification vessel and one in which the crucible is used as the actual solidification vessel. The use of a separate crucible and casting vessel necessitates a pouring operation, which may be accomplished by top pouring or bottom pouring. Each of these techniques has distinct advantages and disadvantages.

4.4.3.1 Top Pouring

A top pour is a physically simple operation that merely requires rotating the crucible to allow the molten metal to pour into the mold or casting vessel. There are several disadvantages to this process. Offgas of the alloy being poured will also have to be addressed for top pouring.

4.4.3.2 Bottom Pouring

Bottom pouring avoids all of these problems but introduces one additional difficulty. For bottom pouring, there are two means of regulating the flow. One of these is the stopper rod, a plunger which is raised or lowered from above the crucible and which opens and closes a hole in the bottom of the crucible. The stopper rod produces good regulation of pour rate and long life of the assembly. The other method is the slide gate, which is a sliding portal that either lines up with or blocks an opening in the bottom of the crucible. The slide gate offers more reliable shutoff than the stopper rod valve but may have a shorter service life. Slide gates are also more expensive and offer less control of the pour rate. In either case there is a potential problem with blockage of the flow regulation device or of the pour spout which is an integral part of the assembly and which guides the flow into the mold. Previous work with aluminum alloys, however, has shown that these alloys may be reliably processed by a bottom pour system with no clogging or flow problems.

4.4.3.3 In-Situ Solidification

The final method of producing a solid SNF form is solidification in crucible. This avoids all of the aforementioned problems and reduces the complexity of the offgas system needed for capture of volatile radioactive species that are released from the melt. Solidification in crucible provides the slowest cooling rate and simplifies the microstructure of the SNF form. It is an inherently safe process because there is no chance of spillage. The lack of melt disturbance (as would be present after pouring) reduces the potential for porosity in the casting and reduces the amount of

volatile species released to the offgas system. Solidification in crucible also allows the slag and oxides to become an integral part of the casting and eliminates the need for separate disposal.

4.4.3.4 Casting Technique Requirements/Recommendations

The basic requirement for the casting technique is the ability to produce a monolithic casting that can be packaged in the SNF canister. The recommended technique is to solidify the melt in the crucible. This relatively simple method allows ease of offgas system and remote handling operations. The integrity of the offgas system will not have to be broken in order to accomplish a pouring operation. Additionally, since no molten metal stream will be passing through a pouring orifice, there is no chance for pour spout clogging/degradation. Proper design and selection of the crucible materials should allow for (1) easy removal of the solidified ingot from the crucible or (2) placement of crucible and ingot in canister for storage, making it compatible with remote handling.

Experimental melts of binary uranium-aluminum alloys have been prepared using tapered graphite crucibles and allowing solidification to occur within the crucible. Crucibles with approximately two-degree taper were fabricated and tested to determine the ability to remove a melt-dilute ingot from a crucible following solidification. These ingots were easily removed from the crucible and little or no skull/slag remained attached to the walls or bottom of the crucible.

4.4.4 Crucible Materials and Batch Size Requirements.

The choice of the appropriate crucible materials depends on several variables including temperatures, melt interaction, and batch size. Several different crucible materials may be viable for operation of the melt-dilute treatment process. The typical commercial foundry crucible materials that warrant consideration are graphite carbon steel, and oxide ceramics.

4.4.4.1 Temperature

As previously stated, the upper temperature limit for the melt-dilute treatment process has been established at 1000°C. Graphite and oxide ceramic crucible can readily handle the 1000°C temperature limit. The carbon steel crucible material is approaching its practical limit at 1000°C as shown in Figure 4.10. Iron concentration versus temperature was plotted from experiments of melting binary U-Al alloys in carbon steel crucibles. From this curve it is evident that at 1000°C there is dissolution of the crucible resulting in higher iron concentrations in the melt. Thus, should the final treatment cycle temperature approach 1000°C, the viability of using cheap, readily available carbon steel crucibles must be questioned.

4.4.4.2 Melt Interactions

In order to limit changes in melt processing chemistry that would hinder the overall process, the interaction between the melt and crucible should be kept to a minimum. The difficulty in this requirement is that aluminum and aluminum alloys tend to be highly chemically active, especially at elevated temperatures. Molten aluminum reduces most other metallic oxides and forms numerous alloys and compounds with the majority of the elements of the periodic table. For the case of graphite crucible, aluminum and some fission products can potentially form carbides, however, the kinetics of this process are slow, and as such, these carbide phases are usually not detected in aluminum alloy castings. In considering ceramic oxide crucible materials, it is important to note that the more thermodynamically stable metallic oxides are magnesium and calcium. However, one consideration with the use of oxide ceramic crucible materials is the ability of molten aluminum and aluminum alloys to wet metallic oxides. For example, although zirconium oxide is thermodynamically less stable than aluminum oxide and thus should be reduced by molten aluminum it is commonly used in commercial casting operations because

molten aluminum does not wet zirconia. Lastly, for the case of using carbon steel crucibles, aluminum form numerous intermetallic phases with iron and would thus appear to be an inadequate choice for a crucible material. However, as depicted in Figure 4.10, the kinetics of the reaction between the two are sluggish enough such that if the time at temperature is minimized carbon steel will not react with the melt.

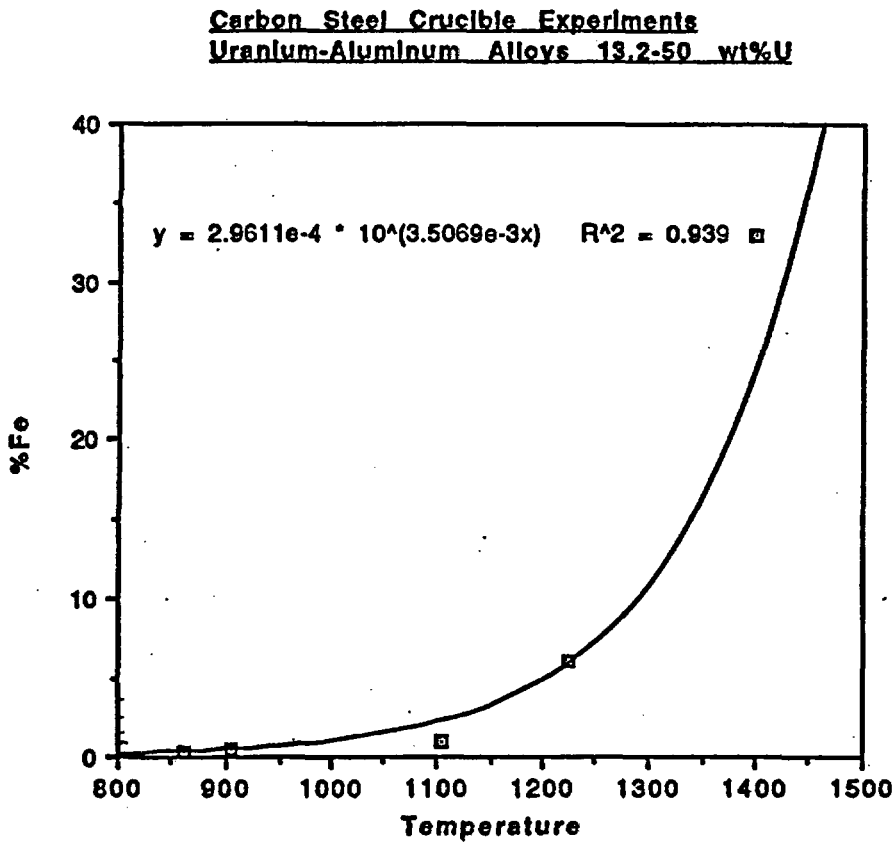


Figure 4.10 Concentration of Iron in U-Al Alloy Melts Following Melting in Carbon Steel Crucibles

4.4.4.3 Batch Size

It is currently envisioned that the crucible for the melt-dilute treatment process should be able to accommodate 15-20 MTR type elements or a single HFIR core. The current revision of the Repository Acceptance Criteria calls for the DOE SNF canister dimension to be on the order of 17 inches in diameter and 120 inches in length. Thus, in order to ensure that the melt-dilute ingot resulting from the treatment process is able to fit into the canister the proposed crucible dimensions for the melt-dilute process are 16.5 inches in diameter by 36 inches. A crucible of this size would have a volume of approximately 135,000 cc and can easily accommodate 15-20 MTR assemblies or a single HFIR core. For the requirement of being able to accommodate one HFIR core the following is necessary owing to the fact that the nominal diameter of a HFIR core is 18 inches these assemblies must be mechanically reduced in size so as to fit into the prescribed crucible. This reduction in size can be accomplished in several different ways—crushing, cutting, etc.—and will be the task of a design engineering team. Additionally, for the pin-type MTR fuel

which are approximately 144 inches in length a mechanically reduction step is also necessary to accommodate these assemblies within the expected crucible and offgas system sizes.

Preliminary shielding calculations for a melt-dilute facility in the process room of the 105-L reactor building for a batch of MTR type fuel have been performed. The dose rate outside the process room wall is slightly above the limit for a continuously occupied area (0.25 mrem/hr). However, the conservatism built into the model provides sufficient margin such that no additional shielding should be required. The model used for these calculations was based on the following assumptions:

1. The fuel assemblies contain 5grams of cesium each
2. Sixteen assemblies will be processed in a batch, with an additional sixteen assemblies drying in the room.
3. The shielded transfer cell has two ingots in process at any one time
4. Dose rate estimates will based on ATR assemblies

4.5 Post-Treatment Characterization

Characterization of the melt-dilute SNF form following treatment results from the need to document the concentrations of approximately 39 radionuclide species. According to the February 1998 Disposability Interface Specification Document Standard 2.3.20.2, information is required on the radionuclide inventories in disposable multi-element canisters delivered to the MGDS. A listing of these radionuclides is provided in Table 4.1 below.

Table 4.1 List of 39 Isotopes to be Characterized as per the DIS

^{227}Ac	^{129}I	^{239}Pu	^{229}Th
^{241}Am	^{93m}Nb	^{240}Pu	^{230}Th
^{242m}Am	^{94}Nb	^{241}Pu	^{232}Th
^{243}Am	^{59}Ni	^{242}Pu	^{233}U
^{14}C	^{63}Ni	^{226}Ra	^{234}U
^{36}Cl	^{237}Np	^{228}Ra	^{235}U
^{244}Cm	^{231}Pa	^{79}Se	^{236}U
^{245}Cm	^{210}Pb	^{151}Sm	^{238}U
^{246}Cm	^{107}Pd	^{126}Sn	^{93}Zr
^{135}Cs	^{238}Pu	^{99}Tc	

Analysis of the remaining portions of the dip samples (≈ ½" X 1") collected for process control sampling is adequate for determining the necessary 39 radionuclide isotopes. Additionally, highly precise numbers concerning total fissile species can be determined during the post-treatment characterization stage if deemed necessary. Unlike, the process control sampling step where time of analysis is crucial –need for quick, accurate response—there is no practical time constraint for the post-characterization step. During this characterization stage, the emphasis is on data precision and quality since it will ultimately be provided for further inputs regarding thermal analysis, dose, and performance assessment. Several commercial analytic techniques are available for providing these high precision values such as, gamma-ray spectroscopy, thermal ionization mass spectroscopy, and x-ray fluorescence spectrometry. Additionally, the possibility exist for the ability to perform bounding measurements and calculations such that each melt-grab sample need not be subjected to extensive analytic scrutiny in order to provide quality data for the variety of repository inputs. Regardless of technique or approach, the melt-dilute treatment

process post-treatment characterization station must be capable of providing precise values for the DIS required 39 radionuclide species and for the total fissile material content (U and Pu). Upon completion of the dip sample analysis, the sample will be incorporated in the next melt batch.

4.6 Furnace Requirements

Preliminary design and definition of the functional performance characteristics of the melt-dilute treatment process facility are focused on the operation of melting irradiated fuel assemblies. This section details the basic requirements for the melt furnace with respect to type, power, stirring, and potential operating mode.

4.6.1 Furnace Type

The two basic furnace types that have been evaluated during the development of this treatment technology are resistance and induction. Small-scale melting facilities have been designed and operated with both furnace types. Initial experiments conducted in the small-scale resistance furnace apparatus (Figure 4.11) showed several disadvantages to using this type of system and lead to the design and construction of the small-scale induction furnace apparatus used in our integrated process demonstration. Two of the disadvantages to using a resistance furnace system are related to melt stirring. Since, there is no built-in stirring capability with a resistance furnace an external mechanical stirrer is needed. This adds complexity to the apparatus as well as mechanical stirring being less efficient than induction stirring. Additionally, removal of the mechanical stirrer presents problem with respect to the spread of contamination. Thus, the recommended furnace for this operation is a standard commercial water cooled induction furnace with a power supply designed to couple with the selected crucible material that is required for this facility. The frequency of the induction power supply will have to be determined by the furnace vendor but is likely to be in the range of 300-1,000 Hz.

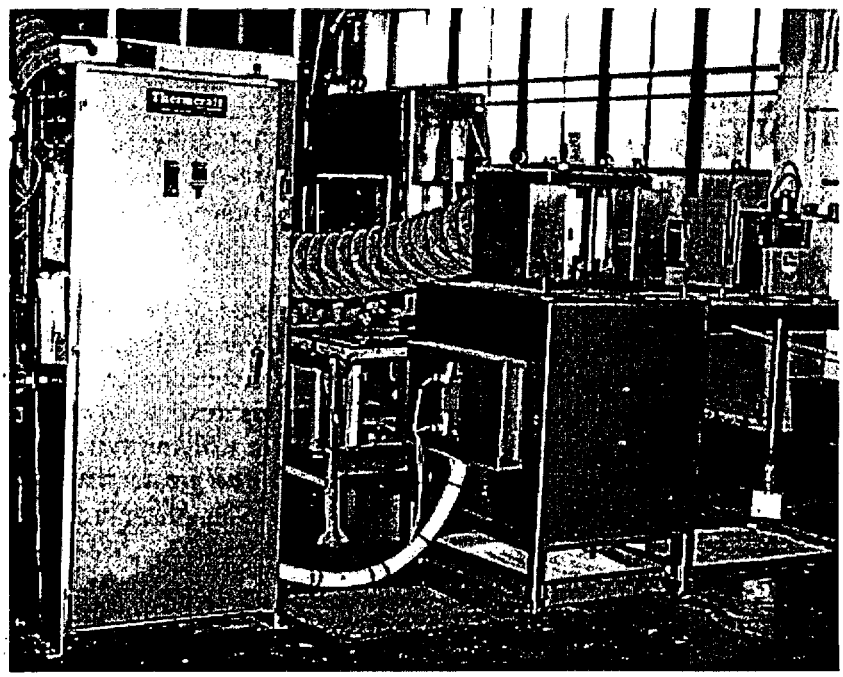


Figure 4.11 Small Scale Resistance Furnace Apparatus

Modern induction furnaces operate at frequencies of 70-5000 Hz, allowing high power densities and producing a natural stirring action in the melt. There are two types of induction furnaces, channel and coreless.

The channel type furnace has a lower chamber that connects to a channel that wraps completely around the primary induction coil. This type of furnace has two inherent drawbacks. Only a small portion of the metal (that inside the channel) receives energy at any one time, limiting the energy transfer. A molten heel must be kept in the lower chamber and channel at all times. If the material completely solidifies, refractory damage will result and restart of the furnace will necessitate a small molten charge to initially fill the channel.

The coreless furnace, which is preferred for this application, uses the crucible and the metal inside as the secondary. The primary wraps completely around the crucible, in the form of a water-cooled copper coil. The advantage of this design is that it may be completely emptied, has high-energy transfer, and is simpler in construction. Another advantage is that the melt is stirred more vigorously because the field may couple with the entire molten volume.

4.6.1.1 Power and Cooling

A primary transformation from the high-voltage grid of the local utility is usually required because the voltage applied to the coil is between 500 and 3000 V. The typical conversion is in an ac-dc-ac sequence.

A closed-loop water system is generally used to cool the furnace coils and power supplies of the induction furnace. In a medium-frequency power supply, the electronic components of the system contain a dc leg; it is therefore important to have clean, demineralized water circulating, typically with conductivity set at 50 mS (50mmho) in order to prevent electrolysis from taking place.

4.6.1.2 Melting Operations

There are two ways to operate a coreless induction furnace. One involves completely emptying the furnace of molten metal each time and recharging it with solids for the next run. The other involves a tap-and-charge operation in which one-third to one-half of the furnace contents are emptied each time, leaving a molten heel in the furnace at all times. There are advantages and disadvantages to each method.

The first method has the advantage that the crucible may be used as the mold, eliminating some of the complexity of the melt-dilute process. There is also a power advantage because medium frequency power supplies produce a flux that couples with the cold charge material more effectively than it does with the molten metal. In addition, this method allows the drying of wet or oily charge material in the furnace, minimizing the need for a drying system. In order to avoid bridging, the length of the solid pieces in the charge should be no greater than two-thirds of the crucible diameter. Longer sections should be cut or sheared to length before insertion into the crucible. This is the preferred method for the melt-dilute process.

The second method has the advantage of producing longer refractory life due to reduced thermal cycling and improved melt chemistry because less of the furnace contents need to be replenished at one time.

4.6.2 Stirring

In addition to an induction power supply required to heat the crucible, an induction stirring power supply will also have to be provided for this facility. The stirring power supply will operate at 60 Hz.

When alternating current is applied to an induction coil, it produces a magnetic field, which in turn generates a current flow through the charge material, heating and finally melting it. The amount of energy absorbed by the charge depends on the magnetic field intensity, electrical resistivity of the charge and the operating frequency. The induced current in the charge creates a second magnetic field. Because these two fields are always in opposite directions, they create a mechanical force that is perpendicular to the lines of flux and cause metal movement or stirring when the charge is liquefied. The mechanical force stays perpendicular to the field only in the center of the coil; on both ends of the coil it changes direction. The metal is pushed away from the coil, moves upward and downward, and flows back. This is referred to as four-quadrant stirring. The stirring action aids in alloy and charge absorption and produces a more chemically and thermally homogeneous melt. The stirring is directly determined by the amount of power induced and is inversely proportional to the square root of the frequency. Therefore, increasing the power while lowering the frequency will result in increased stirring intensity. The two power supplies for the furnace and stirring unit are independent of each other and are designed so they can never operate simultaneously.

4.6.3 Furnace Power

The furnace induction power supply should be designed to deliver the proper level of power to the furnace in order to melt the expected charges of 15-20 MTR assemblies or a single HFIR core. Current understanding of the operation of the melt-dilute treatment process facility indicates that a single melt batch will be run pr day. Thus, the furnace power supply should be able to melt the appropriate size charge and addition materials in a manner such that the entire melt batch procedure can be completed in a single operating shift. A power supply unit in the range of 200-300 kW would be fully capable of meeting these needs.

4.6.4 Atmosphere

The entire melt-dilute process will take place in an air atmosphere. No special cover gas is specified. The inside of the melt crucible will be at a slight negative pressure to minimize the possibility of escape of any evolved gases. In considering atmosphere, it should be noted that graphite at high temperature would react with air to form gaseous oxides of carbon. Thus, should graphite be the crucible material of choice the atmosphere above the melt is likely to be deficient in oxygen, and it may be necessary to supply air if available oxygen is required to react with gasses that are evolved from the melting fuel.

4.7 Offgas System

4.7.1 Background and Overview

The design and development of a furnace offgas system is dependent on basic knowledge about the process used to melt and consolidate the spent nuclear fuel assemblies and the radionuclides contained within the individual assemblies. Fission products are released from the molten fuel and must be captured and contained by an offgas system. A significant amount of basic data is available in the literature on fission product release from molten aluminum-clad, uranium-aluminum alloy fuel. Much of the data has been developed in Severe Accident Analysis studies

involving reactor core and fuel melting scenarios. In addition, computer codes like ORIGEN have been developed to analyze the radionuclides generated in the fuel during reactor operations. An in-depth review of the literature has been conducted to determine the fission products released from the melting of irradiated aluminum-clad, uranium-aluminum alloy spent fuel [7]. In addition, a list of the most important radionuclides generated in spent fuel has been developed from this review.

4.7.2 Radionuclide Inventory for Research Reactor Fuel

Foreign research reactors use a number of different fuel designs [8]. These can be grouped into five categories: (1) plate-type, (2) concentric tube-type, (3) pin-type, (4) special-type, (5) rod-type. The first 4 designs are aluminum-based while the fifth is a TRIGA type. The plate and tube types of (1) and (2) are known as MTR fuels. MTR fuels are used in a majority of the foreign research reactors. For the typical plate type design, the fuel assembly has a box-like housing about four feet in length with two outside housing plates and two outside fuel plates. The number of fuel elements in an assembly varies between 6 and 23, and the initial ^{235}U content varies between 37 gm and 420 gm per assembly. The average burnup of a discharged SNF assembly varies between 15 and 76% (^{235}U percent). The uranium enrichment in MTR type fuel varies from just below 20 to 93 percent.

Radioactive nuclides, including isotopes of the actinides, and fission products are generated during the irradiation of the fuel in the reactor. The ORIGEN computer code [9] can be used to calculate the expected inventory of radioisotopes in research reactor spent fuel. This code has been used extensively at sites throughout the DOE Complex to generate data in support of reactor research and spent fuel analysis. These data have been reviewed at SRS by Bickford and compared to other research reactor fuels to identify important isotopes for use in developing the spent fuel environmental impact statements [10].

A list of some of the most important radionuclides is shown in Table 4.2 below. An extensive list of the radionuclides present in typical MTR assemblies is provided in Table A.9. The list below was developed based on an evaluation of half-life effects, fission yield, and fractional dose calculations for the individual radioisotopes occurring in spent fuel.

Table 4.2 Radionuclides in Research Reactor Spent Fuel

^{3}H (oxide)	^{125}Sb	^{233}U
^{60}Co	^{134}Cs	^{235}U
^{79}Se	^{135}Cs	^{236}U
^{85}Kr	^{137}Cs	^{238}U
^{129}I	^{137m}Ba	^{237}Np
^{131}I	^{144}Ce	^{236}Pu
^{90}Sr	^{147}Pm	^{238}Pu
^{90}Y	^{154}Eu	^{239}Pu
^{95}Zr	^{155}Eu	^{240}Pu
^{95}Nb	^{228}Th	^{241}Pu
^{99}Tc	^{232}Th	^{242}Pu
^{106}Ru	^{233}Pa	^{241}Am
^{110m}Ag	^{232}U	^{243}Am

4.7.3 Offgas System Requirements

4.7.3.1 Radionuclide Species of Interest

A large number of radionuclides with widely varying release characteristics exist in the reactor core at the end of an operating cycle. These fission products may be grouped into classes based on their release characteristics. Severe accident analysis performed for the existing SRS reactors have shown that the thermophysical characteristic of the radionuclides that provides the greatest insight into fission product evolution from the molten fuel is the elemental boiling point.

The fission products can be grouped into five classes based on their boiling points:

Class A: Noble gases and tritium
Boiling point range: < 27°C (300 K)

Class B: Iodine, Cesium, and Rubidium
Boiling Point Range: 27-927°C (300 K to 1200 K)

Class C: Tellurium
Boiling Point: 927°C (1200 K)

Class D: Strontium and Barium
Boiling Point Range: 927-1727°C (1200 K to 2000 K)

Class E: Other radionuclides
Boiling point: > 1727°C (2000 K)

In severe accident analyses, the elements listed above represent the source terms. The elemental boiling points are used as the delimiting characteristic for estimating the magnitude of release, because the primary release mechanism is vaporization of volatile species. The release parameters identified here are time and temperature, or rather, time-at-temperature. Once the boiling point of a species is surpassed, the species will begin to evolve from the molten pool. If the molten pool remains at a given temperature for a long period of time, elements with boiling points below this temperature will eventually be completely released. Additionally, examination of the vapor pressures of the radionuclide species indicates that some level of volatilization can occur at temperatures below the boiling point of the element.

Core meltdown calculations for the HWR-NPR indicated minimal release of radionuclides other than the noble gases from fuel during the assembly melting and in-tank relocation phase (15 minutes from the onset of melting at 650°C). The entire inventory of noble gases is expected to be released at this time. As the molten materials accumulate on the bottom of the reactor tank and heat up to higher temperatures (above 1700°C), substantial quantities of other fission products evolve. Some of the higher boiling point species in Class D are only released to the extent that they are carried by on the gas flows from other isotopes that are more readily released.

Based on the data accumulated from the review of the literature and information on elemental melting point, boiling point, half-lives, and vapor pressure, a listing of the potential radionuclide species that may volatilize at treatment process temperatures in the range of 850-1000°C is shown in Table 4.3 below.

Table 4.3 Elements with Boiling Points Below 1000°C

Element	Symbol	BP (°C)
Tritium	^{3}H	-248
Krypton	^{85}Kr	-153.22
Selenium	^{79}Se	685
Xenon	^{133}Xe	-108.04
Iodine	$^{129,131}I$	184.4
Cesium	$^{134,135,137}Cs$	671
Rubidium	^{137}Rb	688
Tellurium	^{90}Te	988

This list is representative of radionuclides expected, but may not be totally complete as some of the higher boiling point elements may associate with some of the more volatile species and be released at the lower temperatures. Barium (1140°C) and Strontium (1382°C) have been seen coming off in the Argonne experiments above 1050°C.

4.7.3.2 Temperature and Atmosphere Requirements

From the previous discussion of the candidate alloy composition and process cycle options an upper bound temperature for operation of the melt-dilute treatment process of 1000°C has been established. This upper bound temperature is expected to limit the species that will volatilize and thus need capturing in the offgas system. Temperature requirements are such that the system must be able to effectively capture the volatilized fission products. The main radionuclide species of interest is ^{137}Cs due to its relatively high concentration and effective dose. Volatilization of cesium begins at approximately 660°C. According to the literature the fraction of the cesium that can be expected to volatilize can vary from 5-80% depending on temperature and atmosphere. Effective capture of cesium has been demonstrated at temperature from 500°C and below.

Control of the atmosphere within the offgas system is crucial to ensure proper flow of the offgas stream through the filter media. The only two major requirements of the atmosphere are that 1) the atmosphere provided must be able to channel the volatilized radionuclide through the offgas media so as to effect efficient capture of the species, and 2) the system will need to be operated with a slight negative pressure to prevent release from leaks at seals. This will ensure an influx of atmosphere from outside the offgas system and prevent an escape of treatment process gas to the atmosphere. Additionally, the choice of any inert gas backfill is not prescribed.

4.7.4 Offgas System Design

4.7.4.1 Filter Media

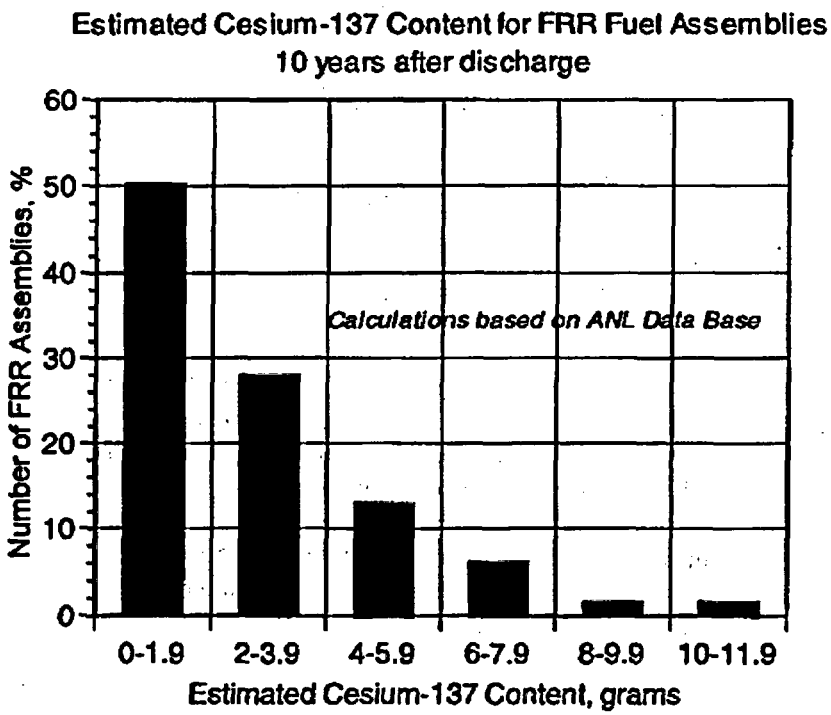
Any furnace design will require an offgas system to capture volatile fission products. The primary contributor to offgas radioactivity will be ^{137}Cs with smaller amounts of ^{85}Kr, ^{99}Tc, and ^{129}I. This section describes some of the available options for design and operation of the offgas system. All the methods considered include some means for condensing the Cs and Tc followed by a HEPA filter to remove particulates. An iodine absorption bed may also be required. We are assuming for the present that the Kr will be released to the atmosphere. If it must be trapped, a silver mordenite bed operated at dry ice temperature will be used. Several means for final disposition of fission products removed from the offgas are being studied.

4.7.4.1.1 Cesium Condenser

The simplest method for removal of cesium from the offgas stream is to cool it to a temperature below its boiling point and to trap the resulting condensate. This will likely be accomplished by use of a metallic labyrinth or other medium on which cesium can be cooled and condensed. The cesium must be kept above its boiling point until it reaches the condenser to prevent its plating out on the surface of the furnace and offgas piping. Possible media for effecting capture of the cesium are glass frit, zeolite, activated alumina, activated charcoal, and crystalline silico-titanate.

Cesium metal melts at 28.5°C and boils at 670°C. The simplest and most effective method for removal of Cs from a hot gas stream is to cool the gas to a temperature well below cesium's boiling point, condensing out the Cs on a cool surface. The Cs can then be disposed of on the condenser medium or be dissolved in an aqueous wash that can then be disposed of to the HLW tanks. The condenser can be located anywhere that its temperature can be kept below 670°C. The most likely sites are the furnace lid or a trap immediately adjacent to the furnace.

A review of the FRR MTR fuel inventory has shown that the concentration of cesium-137 within an individual assembly 10 years after discharge varies from 0 to 12 grams as shown in Figure 4.12. However, approximately 75% of the FRR assemblies contain 4 grams of Cs-137 or less. Thus, bench scale and small-scale experiments have been conducted with cesium doped surrogates based on a nominal Cs-137 concentration of 4 grams per assembly.



Note: Chart does not contain HFIR or RHF data.

Figure 4.12 Estimated Cs-137 for FRR MTR Inventory 10 years After Discharge

Bench-scale laboratory studies with cesium compounds being melted in an aluminum thimble in the apparatus shown in Figure 4.13 as well as full-scale surrogate MTR melts in a small-scale induction apparatus have been conducted. These tests have been conducted using a variety of cesium surrogate materials including cesium metal, cesium oxide and cesium carbonate. Additionally, bench-scale offgas system tests have been conducted in both carbon steel and graphite crucibles to maintain continuity with the treatment process development activities. Analysis of the data from bench –scale offgas test conducted in steel crucibles as shown in Figure 4.14 indicates that regardless of cesium surrogate form very little (≤ 2.5%) is released to the offgas filter system.

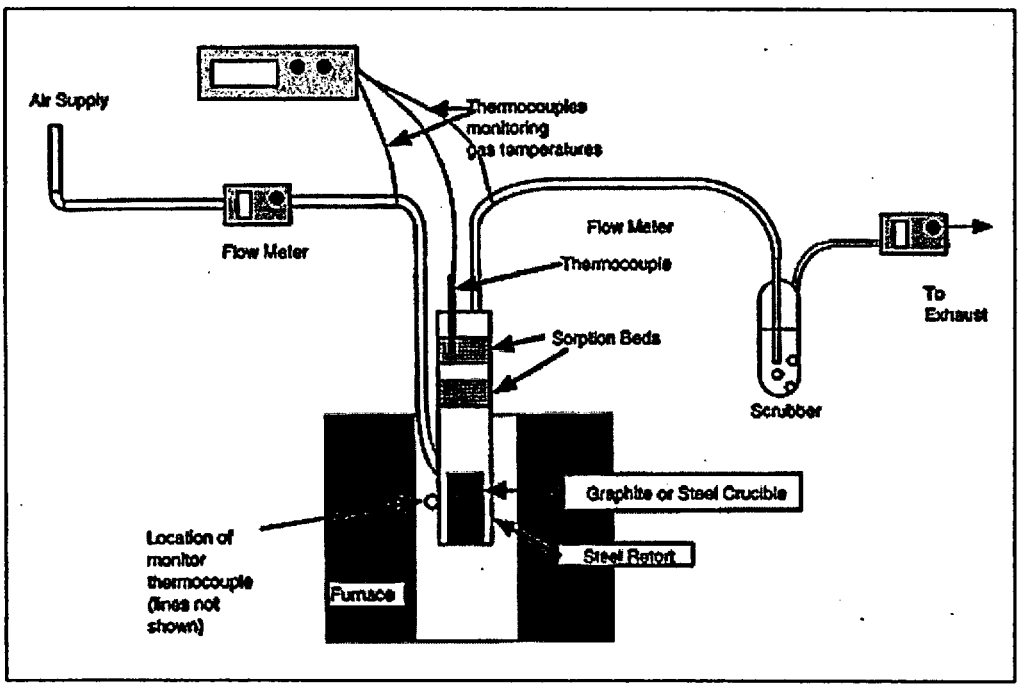


Figure 4.13 Schematic of the Bench-Scale Offgas System Apparatus

Further experiments have been conducted to examine the use of melt-flux materials to inhibit the release of cesium form the U-Al melts. Fluxes consisting of borosilicate glass have been investigated. The borosilicate glass flux is not molten at the expected treatment process temperatures; however it does behave as a viscous fluid and flows to cover the melt surface. Analysis of the melt-flux interface using x-ray fluorescence as seen in Figure 4.15 indicates that the majority of the cesium from the sample is trapped within the borosilicate glass flux and at the flux-melt interface. Thus, further work is in progress to understand the mechanism of flux trapping and to evaluate other candidate flux materials.

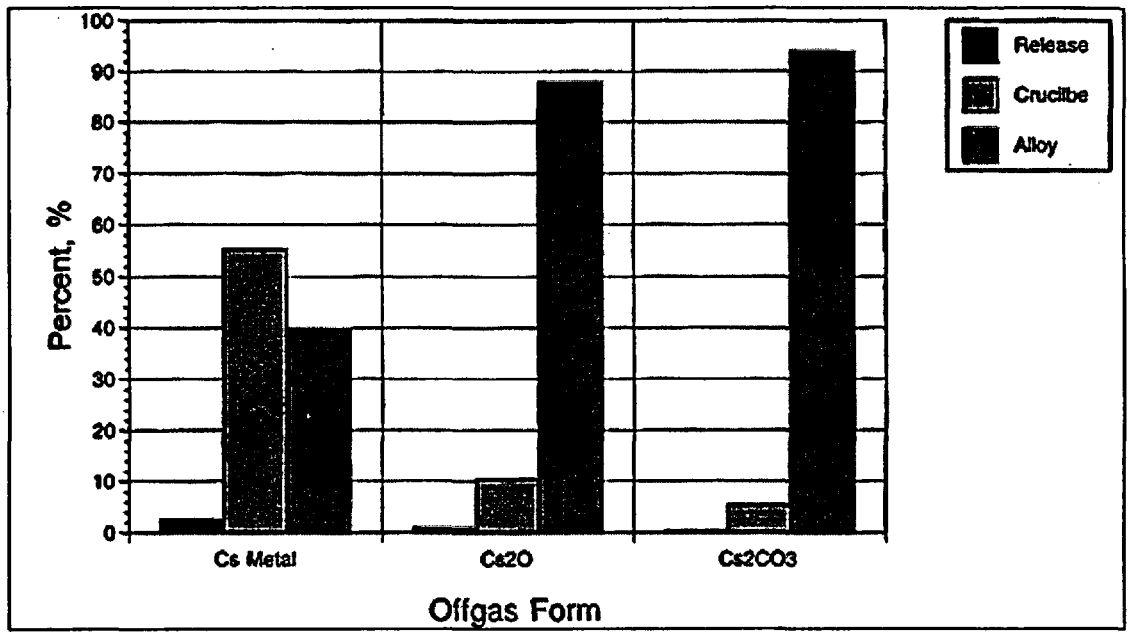


Figure 4.14 Cesium Release from Bench-Scale Carbon Steel Crucible Offgas Melts

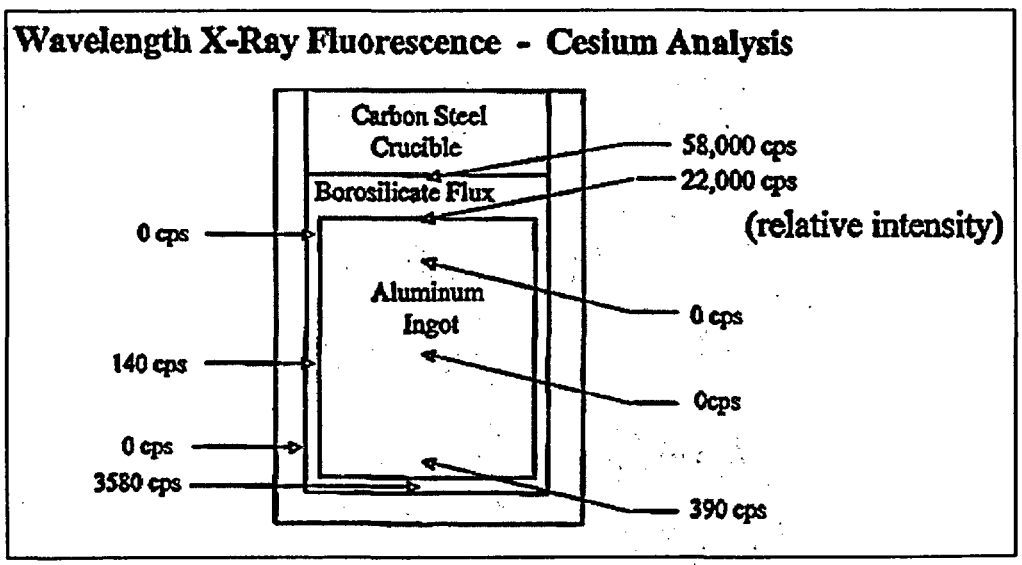


Figure 4.15 XRF Analysis of Al-Billet with Borosilicate Glass Flux

Bench-scale offgas system experiments have resulted in a maximum release of cesium to the filter system of 2.5% as seen in Figure 4.16. Further work has been conducted using the recently constructed small-scale furnace and offgas system (Figure 4.17,Figure 4.18) that is capable of handling a single full-scale MTR fuel assembly. For these tests, surrogate MTR assemblies (Figure 4.19) have been produced using depleted uranium-aluminum alloys as well as various non-radioactive fission product surrogates. Three experiments have been conducted in this system consisting of MTR assemblies doped with varying concentrations of cesium carbonate. The goals of these experiments is to document the effect of scale -up and induction stirring on the

release of cesium from the melt and to address the potential impact on the offgas system design. Preliminary small-scale experiments using three cesium doped surrogate MTR assemblies have produced results consistent with the bench-scale results with respect to the amount of cesium volatilized from the melt and the effective capture of cesium by the dry filter bed.

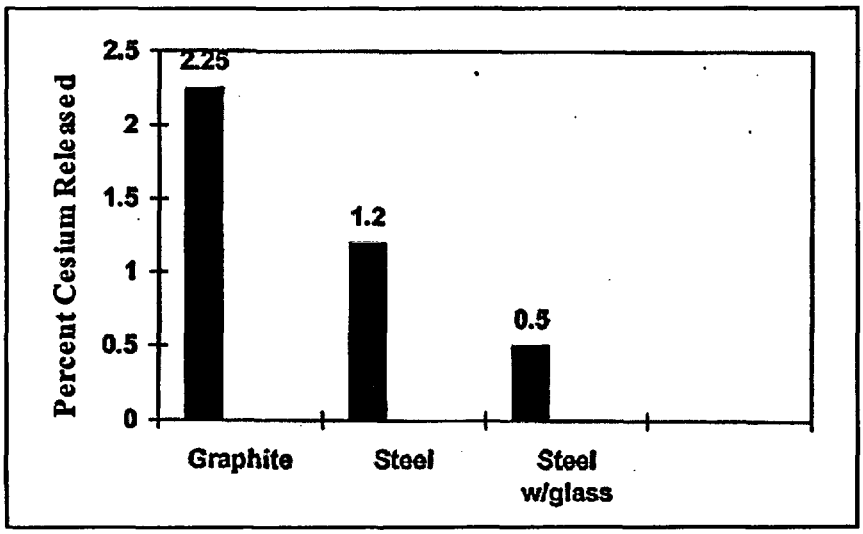


Figure 4.16 Cesium Release to the Offgas Filter in Bench-Scale Offgas Experiments

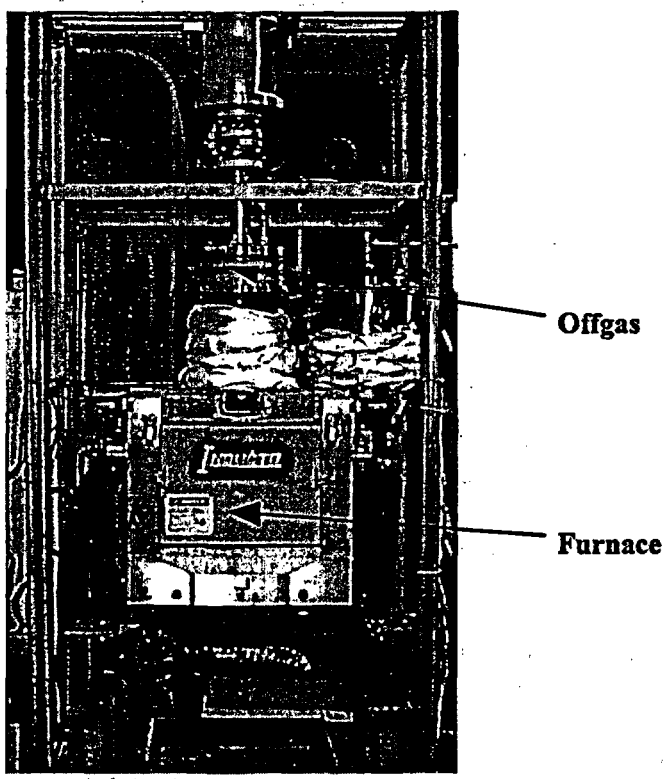


Figure 4.17 Small-Scale Offgas System Photograph

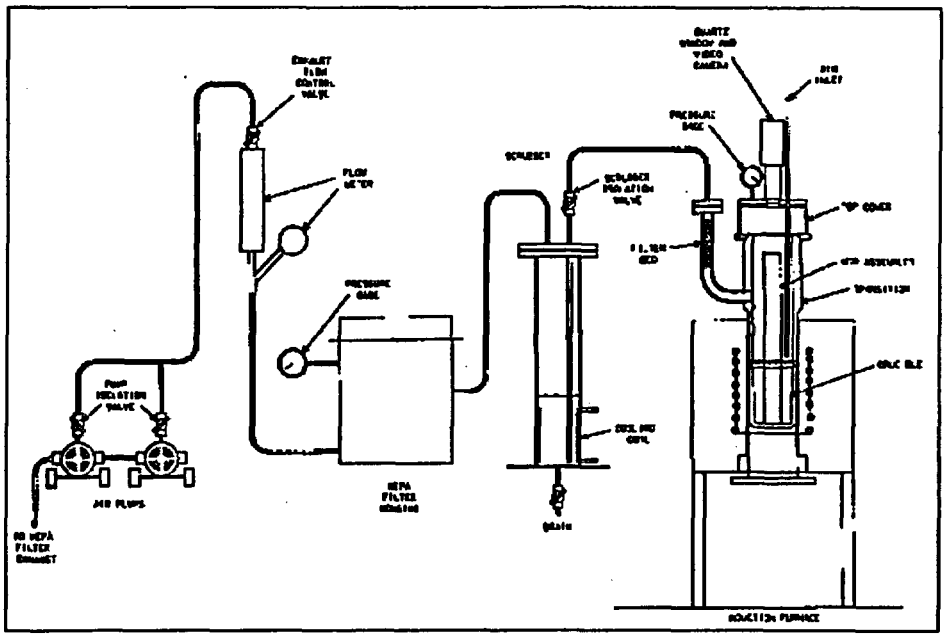


Figure 4.18 Small-Scale Offgas System Schematic

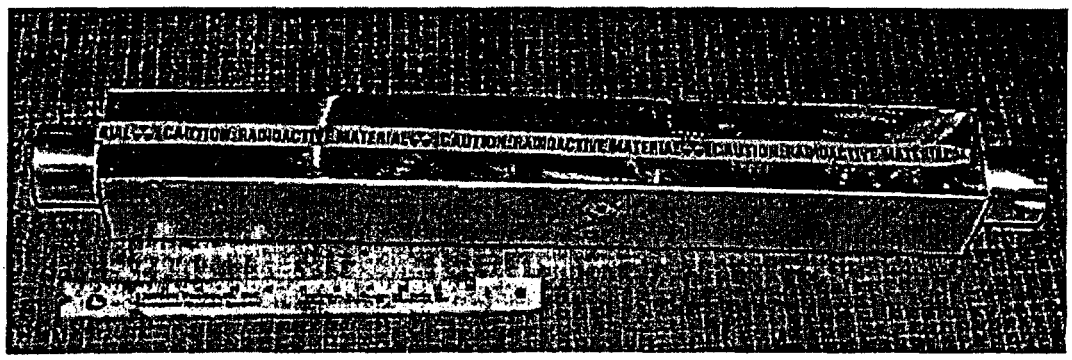


Figure 4.19 Surrogate MTR

4.7.4.1.2 Iodine Reactor (if needed)

The H-canyon process utilizes an iodine reactor consisting of a vessel filled with berl saddles on which silver nitrate has been coated [11]. This reactor is maintained at a temperature between 180 and 200°C. Other designs for iodine reactors have been considered; most incorporating a silver-loaded, heated bed. An activated carbon bed is also effective for I_2 removal. Although any one of these methods could be utilized in the melt-dilute offgas treatment process, the activated carbon bed will probably provide the least expensive option if a temperature of less than about 130°C can be maintained; otherwise a silver-loaded reactor will have to be used.

4.7.4.1.3 Krypton Absorber (if needed)

The issue of krypton absorption is primarily a regulatory one. Preliminary calculations have shown that the radiation dose at the site boundary from the release of this krypton will not exceed 0.04 mrem/year [12], significantly less than the maximum 3 mrem/year dose permitted by 10 CFR 50, Appendix I. Therefore, for planning purposes it is assumed that any Kr released will

be vented to a stack without any attempt at removing it from the offgas stream. Should it become necessary to remove the Kr, a silver mordenite or hydrogen mordenite bed, operated at dry ice temperature, will be effective [13]. The primary difficulty in using the Kr absorber is the necessity to operate it at dry ice temperature with a gas stream originating in a 1000°C or hotter furnace.

4.7.4.1.4 HEPA Filter

No matter what other treatment processes are used to remove fission products from the offgas stream, a HEPA filter will be used to remove any remaining particulate matter before the gas is finally released to the atmosphere via a stack. HEPA filters are available that use either a replaceable glass medium or a permanent (cleanable) stainless steel medium.

4.7.4.2 Flow Rates and System Pressure

The melter system will operate at a slightly negative pressure relative to the atmosphere. The airflow in the bench-scale melter was adjusted to provide approximately three to four air changes per minute (approximately 3.5 scfh). Using the same criterion, the pilot-scale melter will be operated at 3.5 to 5 scfm. To obtain this airflow rate, the initial experiments in the pilot-scale melter will be operated at approximately 5 inches water column. It is expected that the operating pressure in the final system will be in the range of 3-5 inches water column.

4.7.4.3 Monitoring Filter Capacity

In order to ensure effective capture of the volatilized radionuclides and prevent blow-through it is necessary to monitor the capacity of the filter beds. Additionally, safe storage, handling, and transportation of spent filter media are important and as such monitoring activity levels in these media is paramount. Thus, the offgas should be able to monitor the activity levels within the filter beds and offgas media in order to ensure proper operation of the system and also to allow for easy cost effective handling, transportation and disposition of the spent offgas media.

The primary radionuclide of concern is Cs-137. Because it is a gamma emitter, it will be easy to monitor the accumulation of cesium in the sorber material by means of a gamma monitor or ion chamber placed near the filter chamber and shielded from the material in the furnace.

4.7.4.4 Stack Emissions

Stack emissions of I-129, Kr-85, and tritium are predicted to be well below plant release guides and are therefore not considered here. Bench-scale experiments with cesium and aluminum have shown that the cesium released from the melt is less than 2.5% of that charged to the melter. This can be reduced further by the addition of certain materials directly into the melter that trap and immobilize the cesium. The cesium released from the melt will be efficiently trapped on either a zeolite or activated alumina bed. In the bench-scale experiments, essentially no cesium has gotten past the adsorber beds. A HEPA filter will be located downstream of the melter offgas system to catch any particulate material that might get through the sorber bed so that stack emissions will be of little consequence.

4.8 Disposition of Secondary Waste

Selection and design of a system for scrubbing and disposing of cesium from the offgas for the Melt/Dilute operation is expected to be simple and straight forward for the following three reasons. (1) Only a small fraction, < 5%, of the total cesium in the Melt/Dilute operation is transferred to the offgas. Thus, the mass of cesium in the offgas is small. (2) Either a dry or a wet method can scrub cesium from the gas easily and efficiently. (3) There are several simple and practical methods available for final disposal of the cesium. Figure 4.20 presents the

alternate methods, which have been identified for scrubbing and disposing of cesium in the offgas.

Note that the methods divide into two categories: dry or wet scrubbing. The dry scrubbing methods as depicted in Figure 4.21 are preferred and recommended on the basis of fewer process steps and a smaller and simpler HEPA subsystem, since the gas flow is smaller and hot and dry. The dry scrubbing methods must be further evaluated to develop the basis for final selection. The evaluation will include characteristics such as regulatory requirements/permissibility, capital and operating cost estimates, acceptability to the operating division, and acceptability to the organization receiving the cesium.

The simplest system for scrubbing and disposing of the offgas cesium merely places the dry cesium sorbant in the same canister with the melted, diluted fuel rods. The steps of this process are indicated in Figure 4.20. The cesium sorbant, a dry granular material, is held in an aluminum container. The cesium is captured by the sorbant as the offgas flows through it. When the sorbant becomes loaded, the container is removed and added to the storage canister. The offgas will pass through a stack of several sorbant containers so that the bottom container can become fully loaded while the upper containers prevent cesium breakthrough. A fresh sorbant container is added to the top of the stack as a loaded container is removed from the bottom. If necessary, the dry cesium sorbant could be placed in a separate canister by itself.

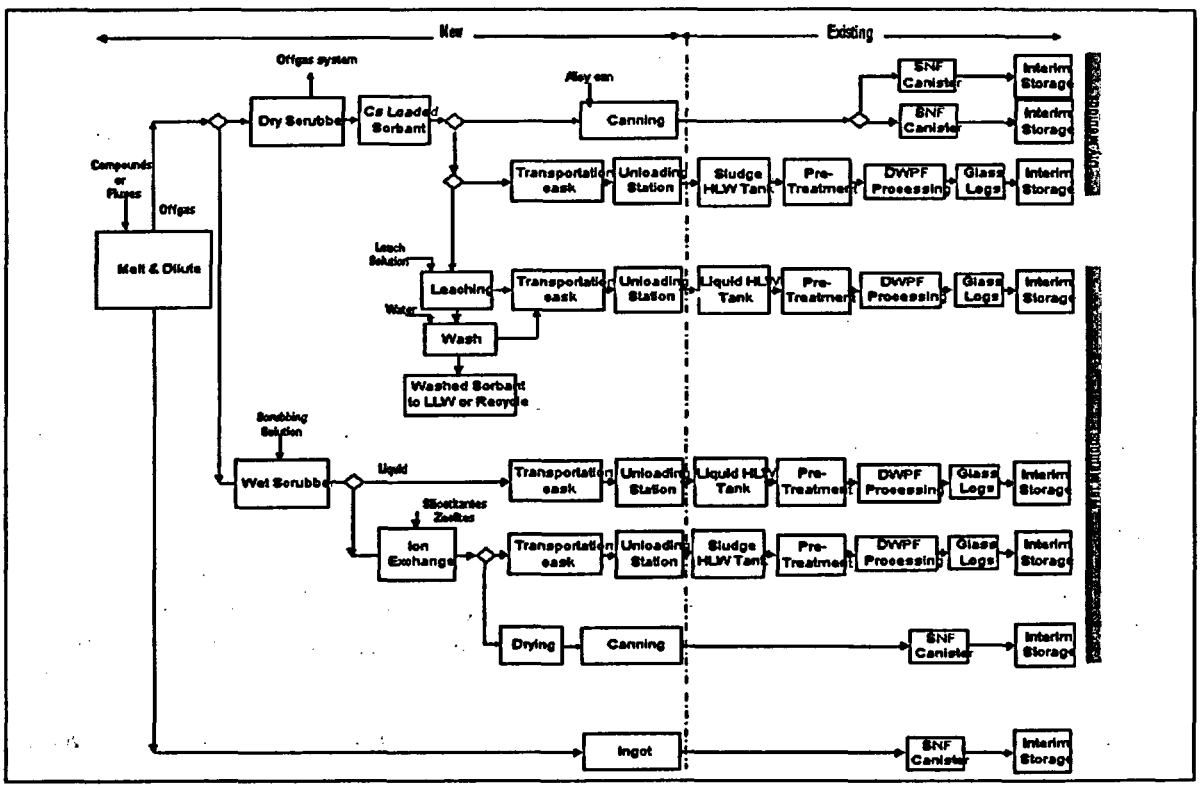


Figure 4.20 Flow Sheet of Melt-Dilute Waste Disposal Treatment Options

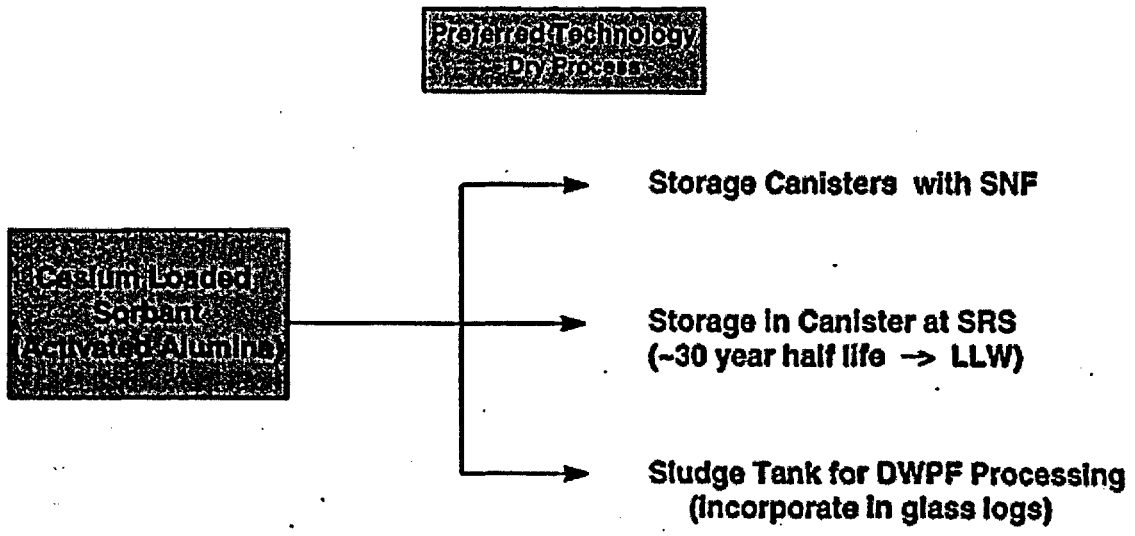


Figure 4.21 Dry Option for Cesium Loaded Filter Bed

Another method for final disposal of the dry cesium sorbant is to combine it with the high level waste going to DWPF. This method requires more steps and handling, but it is practicable. The several possible variants for disposing of the dry cesium sorbant require further evaluation for final selection.

Preliminary shielding calculations for the handling and transportation of the cesium-loaded sorbant via one of the dry methods have been performed. The shielding required to reduce the dose rate at the side f the sorbant canister to below 200 mrem/hr is between 4.5 and 6.75 inches of lead depending on the source (i.e., only cesium or all of the isotopes).

Although the dry scrubbing methods have been selected as the preferred options, several wet methods are also available. The wet methods typically involve a leaching step that results in a high level liquid waste and potentially a low-level solid waste. However, these methods are not preferred because of their additional complexity/processing steps and development requirements.

4.9 Packaging

Packaging considerations for preparation of Al-SNF forms for final disposal in the MGDS include identification and quantification of materials used as neutron absorber materials, backfill gas selection, canister design, and canister labeling. These considerations are addressed in the following sections.

4.9.1 Identification and Quantification of Materials Used as Neutron Absorbers

Analysis of the melt-dilute form within an intact canister (Phase I) [14] has been completed by WSMS, using the KENO-VI computer code, and analysis of the degraded configurations of the melt-dilute form within the waste package is being scheduled. Supporting investigation of poison material compatibility with canister internals, including SNF is in progress. The analysis considered the degradation of the MD SNF form within an intact DOE SNF canister. The results indicate that the product of the MD technology option will require the addition of poison

materials in the event that the DOE SNF canister is not sufficiently filled to exclude moderator to a degree that would preclude the possibility of achieving criticality. The amount of poison materials that would be required to maintain a subcritical condition in the DOE SNF canister will be dependent on the free volume of the canister and on the enrichment. The results indicate that the following parameter values produce a subcritical DOE SNF canister without the addition of poison materials (see Table 4.4). Further poison material requirements will be the result of a Phase II evaluation of the MD SNF form that is planned in FY99.

Table 4.4 Parameters to Preclude Criticality for the MD SNF Form without Poisons

^{235}U Enrichment	Maximum Free Volume (%)
20	10
15	40
10	*

* Enrichment of 10% does not achieve criticality

4.9.2 Backfill Gas

The facility must maintain the capability of adding helium gas backfill to the DOE SNF canister [15]. The canister must be evacuated after loading, backfilled with helium to 1.5 atmospheres, weld-sealed, and leak tested.

4.9.3 Canister Design

The preliminary DOE SNF canister design/construction specifications provided in the recent DIS [15] are summarized herein. The DOE SNF canister shell and lids shall be constructed of a low carbon austenitic stainless steel. The shield plug will be constructed of depleted uranium, sheathed in a compatible material such as stainless steel or carbon steel, or other high density materials except lead, without sheathing if compatible with the basket and shell materials.

DOE SNF canister damage or deformation in the form of bulges, swelling, or dents shall be limited such that the canister continues to meet the dimensional envelope. Dents, holes, and corrosion shall be limited such that the structural integrity of the canister to be lifted and moved intact using remote waste-handling equipment is not adversely affected. Dents, holes, and corrosion shall be limited such that there is no compromise to the canister integrity.

Wastes arriving at the MGDS must be within prescribed size limits in order to be unloaded from casks, handled safely in the surface facilities, and loaded into disposal containers. The two principle reasons for specifying size limits and tolerances are interfaces with waste handling equipment (minimum and maximum reach, dimensions in pool racks, etc.) and compatibility with cavities in disposal containers.

The canisters of DOE-owned SNF that will be place in disposal containers without being opened at the MGDS shall have the capability to stand upright without support on a flat horizontal surface, shall have a diameter no less than 16 inches and no greater than 24 inches (TBV), and shall have an overall length no less than 120 inches and no greater than 180 inches (TBV). The canister shall be capable of fitting without forcing (when lowered vertically) into a right-circular cylindrical cavity whose dimensions are one inch larger than the maximum canister dimension.

Canistered wastes arriving at the MGDS must meet prescribed weight limits in order to be handled safely in the surface facilities, and to ensure that waste packages do not exceed weight limits once canistered wastes are loaded into them. The canisters of DOE-owned SNF that will be placed in disposal containers without being opened at the MGDS shall have a weight that does not exceed 2750 pounds (1247 kg) (TBV). Calculations for postulated loading configurations indicate a maximum canister weight of 3786 pounds for the MD option [16]. Measurement of the weight of the loaded canister should be performed to verify the weight is within the standard for the MGDS.

Measurement of the contact exposure rate of the loaded canister (e.g. at middle of canister) should be performed to allow an assessment of the health physics protection to be taken should the canister need to be retrieved from interim storage. The DIS for the MGDS require the that total heat output of the canister not exceed 1500 W at the time of receipt. Thermal analyses have shown the loaded canister to be well below this value and measurement of the heat output of the loaded canister is not practicable.

Canistered SNF arriving at the MGDS in transportation casks must be unloaded by raising the cask to a vertical position and removing the canisters. Canisters are then either loaded into disposal containers or opened and the contents unloaded. All canisters must be capable of being lifted vertically fully loaded using remotely engaged hooks, grapples or other lifting fixtures at the various stages of unloading, moving, repackaging, and storage. Canisters shall have lifting features that will be accessible and structurally sound to allow lifting vertically with canister-handling equipment.

Disposable canisters delivered to the MGDS shall have lifting features that are compatible with remote handling equipment (cranes, hooks, and grapples) and have the necessary structural integrity to be lifted vertically. These features include a lifting pintle, a standard dimensional interface, and a means to grapple and secure the pintle for lifting. The standard dimensional interface includes a lifting flange and a canister neck with adequate clearance to interface with lifting devices.

4.9.4 Canister Labeling

The facility must maintain the capability of labeling each DOE SNF canister [15] with a unique, permanent alphanumeric identifier. The identifier will be unique to the waste-generator. It shall be visible from the top and side of the canister. It shall be in the form of a label that is an integral part of the canister, can be reasonably expected to remain legible for 100 years at temperatures of 25-400°C, does not impair the integrity of the canister, is chemically compatible with the canister, and does not cause the dimensional limits of the canister to be exceeded. Multi-element disposable canisters shall also have an exterior marking to indicate the SNF form contained within the canister.

4.10 Road-Ready Storage

4.10.1 Interim Dry Storage Requirements

Requirements to limit degradation of Al-SNF throughout the process of receipt, handling, drying, and interim storage are provided in Reference 17. The functional performance requirements listed above in section 4 and below in section 5 of this report embody these requirements. Additional requirements were imposed to provide a road-ready canister and these are contained in

reference 18. For example, the requirement for a helium gas backfill was added to facilitate leak detection in the canisters, to provide improved heat transfer characteristics of the filled canister, consistent with DIS [15] and ICD [19] requirements. Reference 20 demonstrates that these requirements are met for both the melt-dilute and the direct disposal processes as described in this report.

The site requirements for interim dry storage [17] are based on providing for safe, retrievable storage. Retrievability is directly related to limiting degradation in a storage system. In developing the SRS requirements for interim dry storage, limited fuel degradation was acceptable, consistent with the 10 CFR 72 requirement of no gross rupture of fuel cladding during storage and post-storage handling since the fuel was assumed to be eventually removed from interim storage. These criteria were developed prior to the decision to store Al-SNF road-ready at SRS.

The application of "retrievability" for the road-ready storage of Al-SNF forms at SRS is adopted from SRS criteria for interim dry. This achieves both a safety benefit and a potential cost-benefit for the SRS TSF. The safety benefit includes affording safe conditions for postulated events in which the canisters may have failed and the Al-SNF would need to be removed and re-packaged. If repository acceptance would necessitate a direct verification of Al-SNF condition, etc., to meet disposal criteria, a cost benefit is derived from the avoidance of opening canisters following storage at SRS (up to several decades). This is because the characterization information on fuel dimensions, physical condition, etc. assembled before canister sealing, would still be valid at the time of shipment.

Details on the potential degradation of aluminum cladding alloys and fuel materials in drying and dry-storage conditions are provided in References 17 and 21.

Temperature of the fuel must be below 200°C throughout storage in the TSF. This requirement is directly adopted from requirement for direct Al-SNF assemblies [17]. It is recognized that the ingot produced in the melt-dilute treatment does not have the potential creep damage and hydrogen blistering that the direct treatment has and therefore this criterion is conservative.

4.10.2 Monitoring During Canister Storage

It is recommended that the ambient temperature and relative humidity external to the canisters be monitored throughout the dry storage period.

It is recommended that one or both of the following monitoring functions be performed for the canisters throughout the dry storage period:

- Provide for a lead surveillance canister that would be instrumented to measure internal canister environmental and Al-SNF form conditions for the typical storage configuration.
- Implement an inspection program using NDE to evaluate canister and Al-SNF integrity.

4.11 References

1 Savannah River Site Environmental Impact Statement (draft).

2 A. Brewer. "Spent Nuclear Fuel Estimated Quantity of Disposal Canisters", WSRC Calculation Sheet G-CLC-C-0001 (June 1997).

3 G. Long. Explosions of Molten Aluminum in Water–Causes and Prevention, Metal Progress, Volume 5 (1957).

4 P. Hess and K. Brondyke. *Study of Molten Aluminum in Water*, Metal Progress, Volume 4 (1965).

5 A. Lemmon. *Explosions of Molten Aluminum in Water*, Trans. AIME, Volume 4 (1979).

6 D. A. Porter and K. E. Easterling. *Phase Transformations in Metals and Alloys*, Van Nostrand Reinhold, London, pp. 287-290 (1981).

7 J. P. Howell, "Fission Product Release from Spent Nuclear Fuel During Melting (U)," Westinghouse Savannah River Co. Report WSRC-TR-97-0112, May 1997.

8 Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Reach Reactor Spent Nuclear Fuel, Appendix B, Volume 2, DOE/EIS-0218F, February 1996.

9 "ORIGEN 2.1, Isotope Generation and Depletion Code, Matrix Exponential Method," CCC-371, Oak Ridge National Laboratory, Oak Ridge Tennessee, December 1991.

10 W. E. Bickford, "Review of Radionuclides and Source Terms for SRS Spent Fuel," Westinghouse Savannah River Co. Inter-Office Memorandum SPM-SPP-96-0195, Rev. 1, November 6, 1996.

11 K. O'Kula. Personal communication (July 1997).

12 P. R Monson. Krypton Retention on Solid Adsorbents (U). DPST-81-583 (1983).

13 SRS Spent Nuclear Fuel Transfer and Storage Facility Receipt Rate Study. G-ESR-C-000001 (October 29, 1996).

14 S. M. Revolinski. *Criticality Evaluation of DOE SNF Codisposal Canister with Melt and Dilute MTR Fuel.* WSMS-CRT-98-0003 Rev. 0. Westinghouse Safety Management Solutions (February 27, 1998).

15 M. S. Abashian. "Mined Geologic Disposal System Draft Disposability Interface Specification," DI Number B00000000-01717-4600-00108 REV 00. Civilian Radioactive Waste Management System Management & Operating Contractor (February 1998).

16 SFS-FID-98046, "Characteristics of SNF Disposal Canisters," W. S. Large, April 1998.

17 WSRC-TR-95-0347, "Acceptance Criteria for Interim Dry Storage of Aluminum-Alloy Clad Spent Nuclear Fuels (U)," Westinghouse Savannah River Co., March 1996.

18 WSRC-TR-97-00345, "Alternative Aluminum Spent Nuclear Fuel Treatment Technology Development Status Report (U)," Westinghouse Savannah River Co., October 1997.

19 Interface Control Document for U.S. Department of Energy Spent Nuclear Fuel to the Mined Geologic Disposal System for Mechanical and Envelope Interfaces between the Department of Energy Offices of Environmental Management and Civilian Radioactive Waste Management, DI Number A000000000-01717-8100-000007, Rev 00. (December 17, 1997).

20 WSRC-TR-98-00227, "Disposability Assessment: Aluminum-Based Spent Nuclear Fuel Forms (U)," June 1998.

21 R. L. Sindelar, H. B. Peacock, Jr., P. S. Lam, N. C. Iyer, M. R. Louthan, Jr., and J. R. Murphy, "Scientific Basis for Storage Criteria for Interim Dry Storage of Aluminum-Clad Fuels," in proceedings of Scientific Basis for Nuclear Waste Management XIX, Materials Research Society.

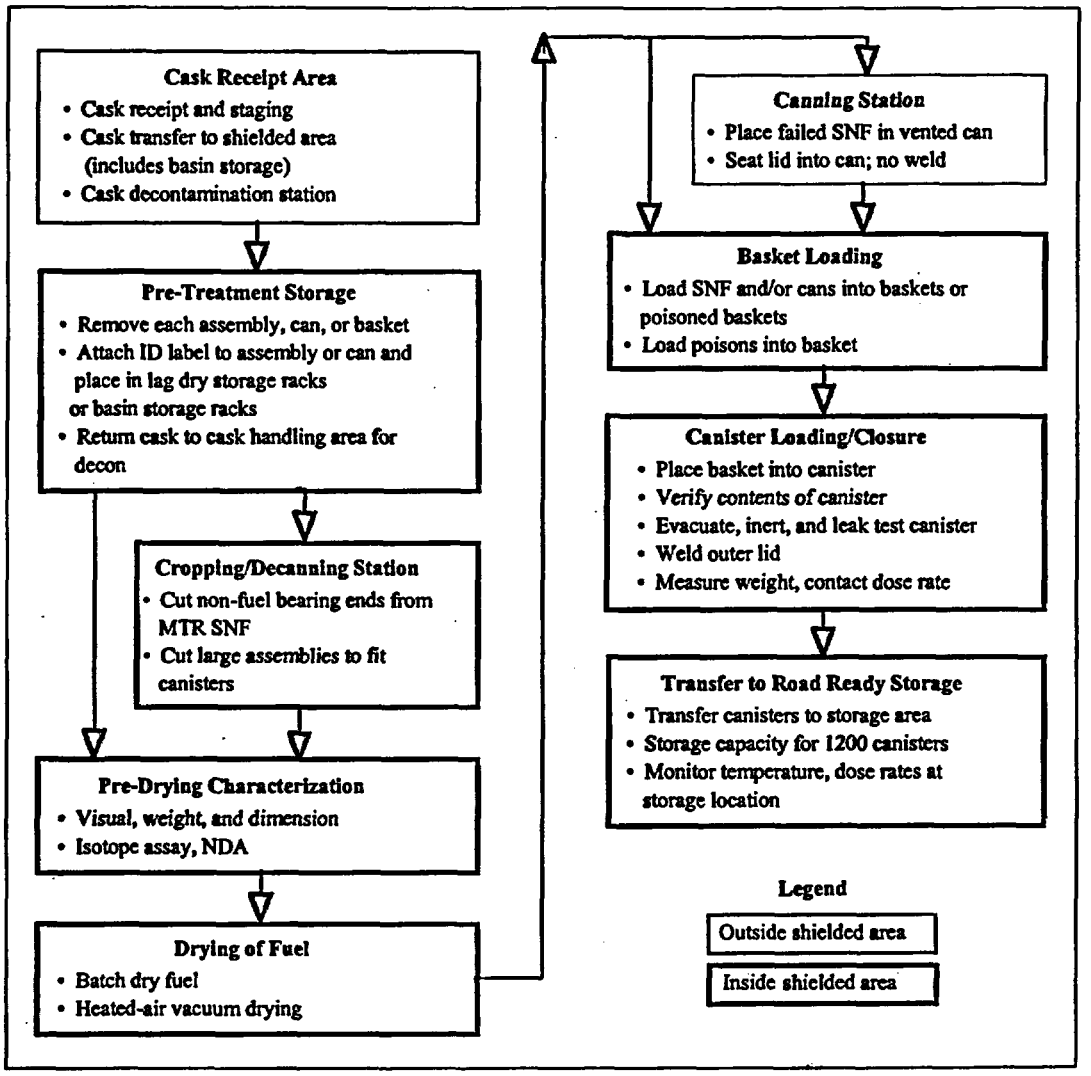


Figure 5.1 Process Steps in TSF for Direct Disposal Technology

For open-air lag storage, it is not feasible to provide capacity for the full inventory of Al-SNF assemblies in the scope (Appendix A). Therefore, using that option, a minimum throughput would be needed based on receipt rate and lag storage capacity.

5.2.1 Pre-Drying Storage Requirements

Al-SNF is presently in basin storage. These are expected to be stored up to an additional one to two decades before treatment. In addition, new receipts will come and be placed in basin storage. Open-air dry lag storage may be used rather than immersing fuel that is received right before treatment. Requirements for basin storage and open-air lag storage are provided.

5.2.2 Basin Storage

The Al-SNF may be stored directly in L-basin within the present acceptance criteria [12]. Existing basin chemistry control should be maintained. The basin standards are listed in the following table.

5.0 DIRECT DISPOSAL PROCESS STEPS AND FPRs

Detailed process steps in the preconceptual design for the direct disposal technology are given in Reference 1. Several process steps are necessary to meet requirements within federal regulations, repository requirements, and site requirements contained in following: 10 CFR 72 [2]; 10 CFR 71 [3]; 10 CFR 60 [4]; DIS [5]; Data Needs [6]; ICD [7]; and SRS Acceptance Criteria [8]. The requirements in References 5 and 8 encompass the applicable requirements except for transportation requirements in Reference 3. Reference 9 demonstrates the viability of both the direct and the melt-dilute Al-SNF form to meet these requirements.

An overview of those process steps that would be needed to prepare the Al-SNF fuel assemblies for road-ready storage to meet the present interpretation of the applicable requirements using direct disposal technology is given by the schematic diagram in Figure 5.1. The steps to be performed within a shielded area would require remote operation or biological shielding to reduce personnel exposure. A discussion of these steps and the associated functional performance requirements for the TSF are provided in this chapter.

The scope of fuels that may be processed using direct disposal are only several of those of the MTR design [10] (see Appendix A, of this report). That is, the involute design and pin design fuels would be limited to processing using melt-dilute technology.

5.1 Al-SNF Receipts at SRS—Shipping Cask Handling, Al-SNF Handling, and Cask Decontamination Functions

No requirements are specified in the interim storage, repository disposal, or treatment regarding the receipt of the shipping casks at SRS and the removal of the Al-SNF assemblies. Standard handling practices and radiological controls at SRS would be used. The FPR for these are covered in Reference 10.

The TSF would not need to provide for or to maintain a shipping cask. Table 8.2-1 in Reference 1 provides physical characteristics of existing transportation casks likely to be utilized to ship fuel to the TSF; however, other transportation casks are expected to be designed, licensed and utilized in the future. The TSF would have to provide for cask return activities including cask decontamination.

Casks needed to transport the DOE SNF canisters containing the Al-SNF forms to the MGDS are anticipated to be within the envelope indicated in Table 8.2-2 in Reference 1.

The projected average fuel receipt rate is 394 Material Test Reactor Equivalent (MTRE) assemblies and 4 High Flux Isotope Reactor/Reactor a-Haut Flux (HFIR/RHF) assemblies per month [11]. Upon receipt, the Al-SNF assemblies would be placed in pre-drying storage.

5.2 Pre-Drying Storage

Two temporary storage locations are envisioned. Basin storage of the assemblies using existing basin storage technology is the first location. The capacity of L-basin is sufficient to store the full inventory of foreign and domestic fuel research reactor assemblies including both present inventory and expected receipts listed as in Appendix A. The second storage option is to provide lag storage for assemblies.

The L-Basin chemistry limits are procedure limits and are contained in Reference 13. The limits and typical values are:

Table 5.1 Table of Basin Chemistry Standards and Present Conditions

Parameter	Limit	Typical Value
Conductivity	10 μmho/cm	1 μmho/cm
pH	5.5 to 8.5	6.2
Chloride	0.1 ppm	< 0.1 ppm (laboratory detection limit)
Cs-137	1000 dpm/ml	13 dpm/ml
Alpha	10 dpm/ml	< 1 dpm/ml (laboratory detection limit)
Tritium	1.0 μCi/ml	0.035 μCi/ml
Al	1.0 ppm	< 0.05 ppm (laboratory detection limit)
Cu	0.1 ppm	< 0.05 ppm(laboratory detection limit)
Fe	1.0 ppm	< 0.05 ppm(laboratory detection limit)
Hg	0.1 ppm	< 0.002 ppm (laboratory detection limit)

Typical values are reported in periodic chemistry reports. The most recent report is Reference 14.

Initial laboratory testing has shown this basin chemistry to be non-aggressive [15]. Basin surveillance confirms that the present storage conditions are non-aggressive [16]. Temporary storage of the Al-SNF in the SRS L-Basin is acceptable. An assessment of the basin storage should be made annually for the stored Al-SNF.

Radionuclide release from exposed fuel meat in water storage is expected to be low [17]. The limit to release from an Al-SNF in water basin storage [12] is 20.7 μCi/hr from an assembly. Fuel can be cropped to expose the fuel meat if the expected release is below 20.7 μCi/hr.

5.2.3 Open-Air Storage

Lag storage of the Al-SNF open to the air is acceptable provided the fuel temperature does not exceed 200°C. Storage racks should be made of aluminum or an insulating material to avoid enhanced corrosion at pooled-water locations. The corrosion of aluminum in air at room temperature is low and an air-storage environment is acceptable [8]. Temperature and relative humidity in the air storage location should be measured, but not necessarily controlled. An assessment of the air-storage conditions should be made annually. Unmonitored storage of Al-SNF in open air should not exceed 1 year.

Argonne National Laboratory has conducted furnace tests for SRTC to evaluate fission product release from irradiated aluminum-based fuel elements. Two heating tests on segments of irradiated aluminum-based fuel have been completed. No radionuclide release was detected from segments of either U_3Si_2-Al or UAl_x-Al during furnace tests at 275°C for times up to four months.

In the first test, a segment of fuel element irradiated in the Oak Ridge reactor was heated at 275°C for 30 days. The fuel was a dispersion of U_3Si_2 (19.8% enriched) particles in an aluminum matrix clad with 6061-T6 aluminum. Average burnup was 51.4%. The area of fuel exposed to air in the test chamber was 0.6 cm^2. In the second test, the fuel element segment was a dispersion of UAl_x particles (19.8% enriched) in aluminum clad with 6061-T6 aluminum. The fuel element

had been irradiated in the Oak Ridge reactor to an average burnup of 66.5%. The area of fuel exposed to air in the test chamber was 0.5 cm^2. There was no release of volatile fission products in either test nor were there any significant changes in fuel microstructure, core-clad interface, or surface oxide thickness detectable by optical microscopy.

5.3 Pre-Drying Characterization

5.3.1 Pre-Drying Characterization Requirements

Federal regulations and repository requirements from 10 CFR 60 and Data Needs and other sources are being input to DIS. The focus of this section is on the specific characterization requirements needed for repository performance and for drying and packaging the Al-SNF. Requirements to ship the Al-SNF were met through the Certificate of Conformance prior to receipt at SRS and it is anticipated no additional characterization of the direct form would be needed to meet these requirements for shipping to the repository.

5.3.1.1 Background

In order to assess the degree of pretreatment characterization necessary to meet both interim dry storage and geologic repository storage requirements, a review of the current regulations and preliminary data needs deemed pertinent to DOE-owned aluminum SNF was performed. Preliminary data needs and system requirements documents for DOE SNF were reviewed along with applicable CFR sections and DOE Orders. A summary of the regulations and requirements that were reviewed is provided in Reference 18, along with specific excerpts and brief interpretations.

By nature, these documents are subject to interpretation, and therefore, the level of applicability to Al-SNF may be subject to interpretation as well. Without firmly established data needs and requirements, however, the Site disposition process must move forward by assuming data needs and requirements from existing documents and governing regulations. The applicability of such regulations and requirements to Al-SNF is therefore being assumed by SRTC/MTS and does not necessarily reflect the interpretation held by other agencies.

The regulations and requirements deemed the most pertinent and applicable to the pretreatment characterization of Al-SNF are summarized in Table 3.1 of Reference 18. The majority of the regulations most applicable to the pretreatment characterization and qualification of the Al-SNF form for repository disposal are under the jurisdiction of 10 CFR, Part 60, "Disposal of High-Level Radioactive Wastes in Geologic Repositories", and 10 CFR, Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste". Specific excerpts from these regulations and requirements are discussed and interpreted in Reference 18, with a complete list of the regulations, data needs, and requirements documents reviewed.

Subsequent to the issuance of Reference 18, the OCRWM issued revision 0 of the waste acceptance criteria, which are the repository requirements for disposal [19]. That document incorporates all previous preliminary versions of waste acceptance criteria and repository performance requirements into a single document. Hereafter in the characterization task, the DIS should be used as the primary reference for requirements, with items identified as "TBD" to be addressed on a case-by-case basis as necessary.

To ensure that the assemblies will fit within the loading baskets with the appropriate clearance and space allowance for poison materials, as necessary, a dimensional check should be performed. The fuel should be slid through a grid to ensure fit into the canister grid; those that do not fit should be set aside for special enlarged contingency grid or for additional cropping or sectioning.

5.4 SNF Drying Treatment

Drying is necessary to meet federal regulations, repository requirements, and site requirements. Two major process options could be used to achieve this: 1) Dry assemblies in batch in drying station and transfer in open air to DOE SNF canister for packaging (option shown in Figure 5.1); or 2) Dry in DOE SNF canister.

5.4.1 Drying Requirements

The site requirement for interim dry storage is based on avoiding hydrogen gas, H_2, build-up during storage in sealed canister. H_2 is generated through corrosion reactions. A limit to the consumption of the cladding material by corrosion throughout handling and storage is imposed (see Section 5.7) to avoid excessive changes in the fuel in the TSF. This requirement is not anticipated to impact the performance requirements of the TSF due to the short times and the low temperature contact with water sources before sealed storage. In a sealed storage system, H_2 build-up gives a more stringent limit for free water in the canister than does corrosion consumption of the material.

Federal regulations and repository requirements specify "no free water" inside the waste package that could compromise the performance of the waste package.

The total amount of water available for corrosion arises from three basic sources. These sources are (1) free water (which includes water in pits in the cladding, crevices, etc.); (2) waters of hydration on existing oxide; and (3) adsorbed waters on oxide and on aluminum. The amount of adsorbed water on the surface of a metal varies with the relative humidity and the temperature. Volpe [20] determined that at 20°C, the amount of absorbed water is approximately 20 monolayers at 100% relative humidity. This is much less than the water of hydration in a 50 μm Boehmite film, assumed to be the initial condition of the cladding, and is therefore negligible.

Drying to a free water level of 9.4 grams per cubic meter of volume of a fuel storage container has been suggested by licensing authorities in Germany [21]. The container could hold 2 MTR elements, and therefore approximately 1.9 grams water/m^2 of cladding surface would be available for corrosion.

Dehydration of the hydrated oxide layers is neither readily achievable nor necessary in drying. The waters of hydration from Boehmite (assumed to be released in storage due to radiolysis) for a film of 50 μm is assumed to be released and to corrode the aluminum cladding to form Al_2O_3. It does not recombine with the original Al_2O_3 left from dehydration of Boehmite. This results in 0.0002 inches of aluminum corrosion. Therefore, the maximum possible uniform consumption of aluminum dried to 1 ml per 0.1 m^2 of cladding surface in a sealed system is approximately 0.0003 inches which is well below the acceptable 0.003 inches of cladding consumption.

5.3.1.2 Characterization Requirements for Al-SNF for Direct Disposal

Each MTR assembly needs to be characterized for isotopic content. Bounding reference assemblies may be constructed for grouping of Al-SNF based on assembly design and reactor histories. No isotopic limit for an individual assembly is given.

5.3.2 Isotope Inventory of Fuel Elements and Assemblies

Characterization of the Al-SNF results from the need to document the amount of 39 radionuclide species. According to the February 1998 Disposability Interface System Document Standard 2.3.20.2, information is required on the radionuclide inventories in disposable multi-element canisters delivered to the MGDS. A listing of these radionuclides is provided in the Table 5.2 below.

Table 5.2 List of 39 Isotopes to be Characterized as per the DIS

^{227}Ac	^{129}I	^{239}Pu	^{229}Th
^{241}Am	^{93m}Nb	^{240}Pu	^{230}Th
^{242m}Am	^{94}Nb	^{241}Pu	^{232}Th
^{243}Am	^{59}Ni	^{242}Pu	^{233}U
^{14}C	^{63}Ni	^{226}Ra	^{234}U
^{36}Cl	^{237}Np	^{228}Ra	^{235}U
^{244}Cm	^{231}Pa	^{79}Se	^{236}U
^{245}Cm	^{210}Pb	^{151}Sm	^{238}U
^{246}Cm	^{107}Pd	^{126}Sn	^{93}Zr
^{135}Cs	^{238}Pu	^{99}Tc	

The required accuracy is provided in the Data Needs report [6]. The Data Needs report [6] also provides guidance to an alternative to directly providing this isotopic information. The initial enrichment, irradiation history and burnup can be provided instead, although this alternative is not provided in the February 1998 draft of the DIS [5].

These isotope quantities can be provided by a combination of a non-destructive assay and analysis. A detection system capability of detecting and quantifying, as a minimum, fission products ^{137}Cs and ^{241}Am and fissile species ^{235}U and ^{239}Pu should be available in the TSF. The balance of the isotopes would be assembled through analysis. Other isotope combinations can be used. To provide this information for the direct Al-SNF form, the TSF would have to have the capability of assaying the fission product and fissile species of individual Al-SNF assemblies using an NDA system. The facility should also provide a data analysis strategy to yield a complete set of the quantities of the above isotopes on a per assembly basis within the required accuracy. This may include construction of a bounding reference assembly for the grouping of Al-SNF assemblies.

5.3.3 Assembly Dimension Verification

A photographic record should be taken of each assembly as a permanent facility record. The assemblies should be dimensionally verified in the pre-drying characterization stage to ensure adequate clearance in the drying station and in the canister. Those assemblies that do not meet the clearances in the drying station and the canister basket would need to be re-cropped or a larger size basket would need to be available as a contingency.

The limiting drying criterion is that to avoid hydrogen build-up. At temperatures above approximately 80°C, hydrogen build-up occurs in a closed system containing aluminum and water according to [22]:

$$2Al + 4H_2O = Al_2O_3 \cdot H_2O + 3H_2$$

Assuming the reaction goes to completion, 1 ml of H_2O yields 0.042 moles of H_2. The generation of hydrogen in the reaction to produce boehmite (above) bounds that for the reaction to produce gibbsite at temperatures below approximately 80°C.

A general formula can be derived to relate the volume fraction of free water to the hydrogen pressure for the reaction at completion [22]:

$$\frac{FW}{V} = 292505 \cdot \frac{P_{H_2}}{(273.15 + T)}$$

where FW is the free water volume in ml;
V is the volume of the container in m^3;
P_{H_2} is the pressure due to H_2 in atmospheres;
and T is the temperature in °C.

One impact of hydrogen buildup is the potential for an explosion hazard. The lower concentration limit of flammability of hydrogen is 4 percent by volume in air at room temperature. The lower concentration level for a sustained burn of hydrogen in air is approximately 9 percent by volume in air. The concentration level at which a hydrogen/air mixture is explosive is 18 percent. Therefore, the partial pressure of hydrogen must be below 0.59 psia or the ratio of free water (ml) to the volume of the container (m^3) must be kept below 39 to ensure that hydrogen at 4 volume percent of a container with air is not produced.

Another impact of hydrogen buildup is the potential for production of UH_3, a compound that is pyrophoric under certain conditions. Dispersoids such as UAl_x would not be reduced by expected hydrogen pressures to produce UH_3; however, oxides of uranium could be if the partial pressure of H_2O is low enough and the partial pressure of H_2 is high enough in an H_2O/H_2 system. However, because aluminum surrounds the oxides in the dispersoid fuel and does not allow direct contact with the fuel particles, this is not expected to result in significant production of UH_3.

Considering the potential for existing UH_3 on uranium metal fuels retrieved from basin storage, only uranium metal fuels may need to be stabilized. The INEL has developed a stabilization treatment with the technical bases [23] to convert UH_3 to uranium oxide. No significant amount of UH_3 is expected to be present on aluminum-based fuels. In addition, the stringent drying requirements ensure that the expected H_2 build-up is extremely low so that UH_3 production due to H_2 gas contacting exposed fuel meat is negligible.

5.4.2 Dryness Specification

To limit the H_2 build-up in a sealed canister to 4% by volume from free water, the maximum allowable free water (W_{water}, in grams) is expressed as a function of the free volume (V, cm^3) of the container.

$$W_{water} = 3.873 \times 10^{-5} V$$

5.4.3 Vacuum Drying Specification

The Al-SNF must be vacuum dried to at least 5 torr at an internal chamber temperature of 25°C (77°F) or higher. The vapor pressure of water at 25°C (77°F) is 24 torr, and any free liquid water on the fuel will be vaporized before 5 torr is reached [24]. To ensure dryness, the vacuum pump should be isolated following drying and the chamber interior pressure must remain at or below 5 torr for 15 minutes to confirm dryness.

Drying tests using unheated vacuuming to dry an instrumented canister containing residual free water and a mock fuel assembly were reported in reference 25. These tests were successful at drying the assembly and canister, and showed that temperature, pressure, or relative humidity could be used as measures of free water removal. Applying a warm air purge during the vacuum drying improved the drying method. Recent field experience with two instrumented, shielded SNF test canisters indicates a continuous warm air purge (< 74°C (165°F)) under vacuum provides satisfactory dryness in a reasonable time interval. A canister containing one assembly and ~0.6 pints of water was dried to ~0.25 torr in 2.5 hours.

The vacuum pump should have a pumping speed of at least 100 cfm and a rated ultimate vacuum of at least 0.5 torr. An air-cooled vacuum pump is recommended to eliminate the need for cooling water disposal with a water-cooled pump. A water-sealed vacuum pump is recommended to eliminate the need for waste oil disposal with an oil-sealed pump. Redundant vacuum sensors should be provided as close as practical to the canister vacuum nozzle connection.

The warm air purge system should have a thermostatically controlled heater and a flow of at least 25 cfm. Temperature monitoring must be provided for air entering and exiting the canister. Direct temperature monitoring of the canister bottom is desired.

The drying temperature shall not exceed 250°C to avoid the potential for hydrogen blistering and gross cladding failure. The drying specification calls for a low-temperature heated-air vacuum drying. The bases for a low temperature during the drying process is to avoid the potential for blistering of the Al-SNF materials caused by H_2 via vapor corrosion at high temperature (at and above 250°C). Additional information on the hydrogen blistering of aluminum is contained in References 8, 22, and 25.

5.5 Post-Treatment Characterization

No characterization of the individual assemblies is required following the drying treatment. Radiation field measurements of the filled DOE SNF canister will be required as discussed in Section 5.6.

5.6 Packaging

Packaging considerations for preparation of Al-SNF forms for final disposal in the MGDS include identification and quantification of materials used as neutron absorber materials, backfill gas selection, canister design, and canister labeling. These considerations are addressed in the following sections.

5.6.1 Identification and Quantification of Materials Used as Neutron Absorbers

Analyses of the Al-SNF direct form within an intact DOE SNF canister (Phase I) [26] and degraded configurations within the waste package (Phase II) [27] have been completed by CRWMS, using the MCNP4A computer code. The Phase I analysis included both intact assemblies and assemblies at different stages of degradation. Results of this analysis indicate that

neutron-absorbing materials are necessary to preclude the possibility of a criticality event. The amount of boron in the borated stainless steel basket required to preclude criticality is 1.37 kg. The Phase II analysis considers the degradation of SNF and structures internal to the codisposal waste package including HLW, canisters, and criticality control material. The analysis indicates that gadolinium phosphate would provide better long-term criticality control than boron, based on geochemistry considerations. Further, due to the higher iron content of carbon steel and to the neutron absorbing characteristics of iron, the analysts suggest that the basket structure be constructed of carbon steel. Results of the Phase II criticality analysis indicates that the most reactive condition occurs for intact basket configurations with degraded fuel. This configuration requires the highest poison content to remain subcritical. The amount of gadolinium that is required in the basket structure to preclude criticality is 1 kg, assuming a stainless steel basket. If carbon steel is used for fabrication of the basket, 1.25 kg of gadolinium is required.

5.6.2 Backfill Gas

The facility must maintain the capability of adding helium gas backfill to the DOE SNF canister [5]. The canister must be evacuated after loading, backfilled with helium to 1.5 atmospheres, weld-sealed, and leak tested.

5.6.3 Canister Design

The DOE SNF canister design/construction specifications are provided in the DIS [5] and are summarized herein. The DOE SNF canister shell and lids shall be constructed of a low carbon austenitic stainless steel. The shield plug will be constructed of depleted uranium, sheathed in a compatible material such as stainless steel or carbon steel, or other high density materials except lead, without sheathing if compatible with the basket and shell materials.

DOE SNF canister damage or deformation in the form of bulges, swelling, or dents shall be limited such that the canister continues to meet the dimensional envelope. Dents, holes, and corrosion shall be limited such that the structural integrity of the canister to be lifted and moved intact using remote waste-handling equipment is not adversely affected. Dents, holes, and corrosion shall be limited such that there is no compromise to the canister integrity.

Wastes arriving at the MGDS must be within prescribed size limits in order to be unloaded from casks, handled safely in the surface facilities, and loaded into disposal containers. The two principle reasons for specifying size limits and tolerances are interfaces with waste handling equipment (minimum and maximum reach, dimensions in pool racks, etc.) and compatibility with cavities in disposal containers.

The canisters of DOE-owned SNF that will be place in disposal containers without being opened at the MGDS shall have the capability to stand upright without support on a flat horizontal surface, shall have a diameter no less than 16 inches and no greater than 24 inches (TBV), and shall have an overall length no less than 120 inches and no greater than 180 inches (TBV). The canister shall be capable of fitting without forcing (when lowered vertically) into a right-circular cylindrical cavity whose dimensions are one inch larger than the maximum canister dimension.

Canistered wastes arriving at the MGDS must meet prescribed weight limits in order to be handled safely in the surface facilities, and to ensure that waste packages do not exceed weight limits once canistered wastes are loaded into them. The canisters of DOE-owned SNF that will be placed in disposal containers without being opened at the MGDS shall have a weight that does not exceed 2750 pounds (1247 kg) (TBV). Preliminary calculations by SFSD indicate a

maximum canister weight of 3006 pounds for the DD SNF technology option [28]. Measurement of the weight of the loaded canister should be performed to verify the weight is within the standard for the MGDS.

Measurement of the contact exposure rate of the loaded canister (e.g. at middle of canister) should be performed to allow an assessment of the health physics protection to be taken should the canister need to be retrieved from interim storage. No requirements are specified for the measurement of the exposure rate from individual assemblies. However, measurement of the exposure rate from individual assemblies can be readily made during the isotopic measurements using NDA as discussed in Section 5.3.2.

The DIS for the MGDS require that the total heat output of the canister not exceed 1500 W at the time of receipt. Thermal analyses have shown the loaded canister to be well below this value and measurement of the heat output of the loaded canister is not practicable.

Canistered SNF arriving at the MGDS in transportation casks must be unloaded by raising the cask to a vertical position and removing the canisters. Canisters are then either loaded into disposal containers or opened and the contents unloaded. All canisters must be capable of being lifted vertically fully loaded using remotely engaged hooks, grapples or other lifting fixtures at the various stages of unloading, moving, repackaging, and storage. Canisters shall have lifting features that will be accessible and structurally sound to allow lifting vertically with canister-handling equipment.

Disposable canisters delivered to the MGDS shall have lifting features that are compatible with remote handling equipment (cranes, hooks, and grapples) and have the necessary structural integrity to be lifted vertically. These features include a lifting pintle, a standard dimensional interface, and a means to grapple and secure the pintle for lifting. The standard dimensional interface includes a lifting flange and a canister neck with adequate clearance to interface with lifting devices.

5.6.4 Canister Labeling

The facility must maintain the capability of labeling each DOE SNF canister [5] with a unique, permanent alphanumeric identifier. The identifier will be unique to the waste-generator. It shall be visible from the top and side of the canister. It shall be in the form of a label that is an integral part of the canister, can be reasonably expected to remain legible for 100 years at temperatures of 25-400°C, does not impair the integrity of the canister, is chemically compatible with the canister, and does not cause the dimensional limits of the canister to be exceeded. Multi-element disposable canisters shall also have an exterior marking to indicate the orientation of the SNF basket within the canister.

5.7 Road-Ready Storage

THE REQUIREMENTS ARE IN SECTION 4.10.

The recommendations in Section 4.10.2 include provisions for a lead surveillance canister that would be instrumented to measure internal canister environmental and Al-SNF form conditions for the typical storage configuration. A comprehensive test system in which Al-SNF is loaded into a canister, vacuum-dried, backfilled with a known environment and monitored for fuel response is being performed in the L dry cave in L disassembly. This system may be used as a prototype for a surveillance canister to perform the recommended monitoring function for the TSF. This test canister is instrumented with thermocouples and sensors for temperature, pressure,

and volatilization studies. This monitoring system is networked such that data acquisition can be performed remote form the L-area dry cave. Currently, two damaged fuel assemblies—one with mechanical damage and one with corrosion damage (through-clad pit)-- have been placed in the instrumented canister and are under surveillance. Figure 5.2 shows the 2 instrumented canisters in L dry cave being used for validation testing of drying and storage criteria.

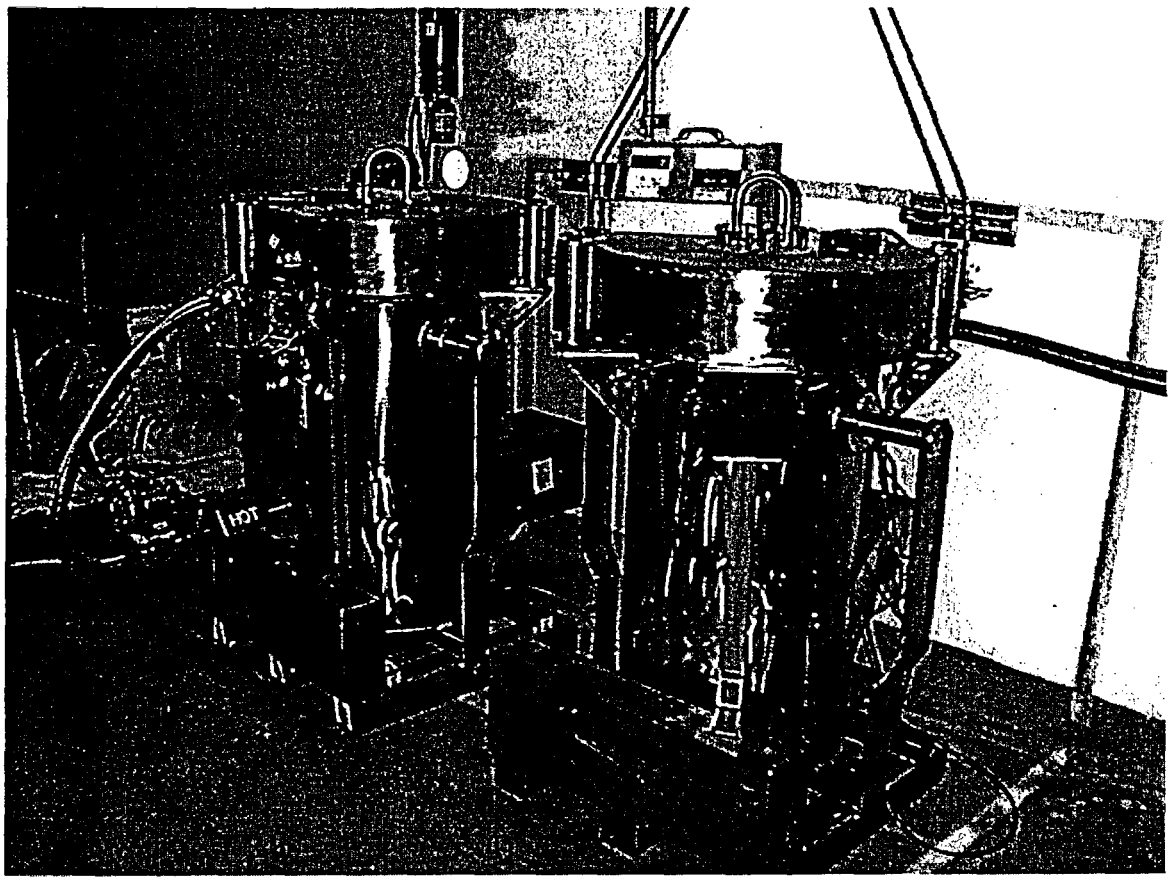


Figure 5.2 Instrumented, Shielded Test Canisters for Validation Testing

5.8 References

1 Savannah River Site Spent Nuclear Fuel Transfer and Storage Services Design, DI Number G-CDP-G-00002. (June 1997).

2 U. S. Code of Federal Regulations, Chapter 10, Part 72.

3 U. S. Code of Federal Regulations, Chapter 10, Part 71.

4 U. S. Code of Federal Regulations, Chapter 10, Part 60.

5 M. S. Abashian. "Mined Geologic Disposal System Draft Disposability Interface Specification," DI Number B00000000-01717-4600-00108 REV 00. Civilian Radioactive Waste Management System Management & Operating Contractor (February 1998).

6 *OCRWM Date Needs for DOE Spent Nuclear Fuel*, DI Number A000000000-01717-2200-00090, Rev 02. (January 1998).

7 Interface Control Document for U.S. Department of Energy Spent Nuclear Fuel to the Mined Geologic Disposal System for Mechanical and Envelope Interfaces between the Department of Energy Offices of Environmental Management and Civilian Radioactive Waste Management, DI Number A000000000-01717-8100-000007, Rev 00. (December 17, 1997).

8 WSRC-TR-95-0347, "Acceptance Criteria for Interim Dry Storage of Aluminum-Alloy Clad Spent Nuclear Fuels (U)," Westinghouse Savannah River Co., March 1996..

9 WSRC-TR-98-00227, Disposability Assessment: Aluminum-Based Spent Nuclear Fuel Forms, June 1998

10 WSRC-TR-98-00218. Functional Performance Requirements for a Proposed Treatment and Storage Facility for Spent Nuclear Fuel, July 1998.

11 SRS Spent Nuclear Fuel Transfer & Storage Facility Receipt Rate Study, G-ESR-C-00001, October 29, 1996.

12 SFS-RSE-970167. "Proposed New Criteria for Acceptance of Al-SNF for SRS Basin Storage (U)," S. D. Burke to R. J. Skwarek by H. M. Brooks, February 11, 1998.

13 Procedure SOP-SAMP-002

14 WSRC-RP-97-231-010

15 "Evaluation of Water Chemistry on the Pitting Susceptibility of Aluminum," G. T. Chandler, R. L. Sindelar, and P. S. Lam, National Association of Corrosion Engineers International, Houston, TX, Paper No. 104 at CORROSION/97, March 1997.

16 "Corrosion Surveillance in Spent Fuel Storage Pools," National Association of Corrosion Engineers International, Houston, TX, NACE'97, Paper 107, March 1997.

17 WSRC-TR-97-0153, "Radioactivity Release from Aluminum-Based Spent Nuclear Fuel in Basin Storage (U)," R. L. Sindelar and J. P. Howell, May 1997.

18 T. E. Skidmore. Pretreatment Characterization Requirements for the Disposition of U.S. DOE Aluminum SNF. WSRC-TR-97-00290 (September 1997).

19 M. S. Abashian. *Mined Geologic Disposal System Waste Acceptance Criteria.* B00000000-0171704600-00095 REV00 (September 1997).

20 L. Volpe, in the Proceedings of the 10th International Congress on Metallic Corrosion, 1987.

21 Letter transmittal, Horst W. Hassel to H. B. Peacock, Jr, March 8, 1996.

22 WSRC-TR-97-0120, Vapor Corrosion of Aluminum Clad and Fuel Materials, P. S. Lam, R. L. Sindelar, and H. B. Peacock, April 1997.

23 M. A. Ebner, "The Potential Pyrophoricity of BMI-SPEC and Aluminum Plate Spent Fuels Retrieved From Underwater Storage," INEL report, 1995.

24 WSRC-TR-97-0075, "Review of Drying Methods for Spent Nuclear Fuel," W. S. Large and R. L. Sindelar, Westinghouse Savannah River Co., April 1997.

25 WSRC-TR-97-00345, "Alternative Aluminum Spent Nuclear Fuel Treatment Technology Development Status Report (U)," Westinghouse Savannah River Co., October 1997..

26 A. H. Wells and J. W. Davis. *Evaluation of Codisposal Viability for Aluminum-Clad DOE-Owned Spent Fuel: Phase 1; Intact Codisposal Canister*. DI Number BBA000000-01717-5705-00011 REV 00, Civilian Radioactive Waste Management System Management & Operating Contractor (June 9, 1997).

27 J. W. Davis and P. Gottlieb. Evaluation of Codisposal Viability for Aluminum-Clad DOE-Owned Spent Fuel: Phase 2; Degraded Codisposal Waste Package Internal Criticality. DI Number BBA000000-01717-5705-00017 REV 00, Civilian Radioactive Waste Management System Management & Operating Contractor (December 15, 1997).

28 SFS-FID-98046, "Characteristics of SNF Disposal Canisters," W. S. Large, April 1998

APPENDIX A

A.0 SCOPE OF FUEL AT SRS TO BE DISPOSITIONED

The SRS is presently consolidating the DOE Al-SNF from foreign and domestic research reactors. A description of these fuels using the best available information is provided in this appendix. The treatment of Al-SNF for ultimate disposition must deal with these fuels. Additional information on these fuels will be obtained through characterizations as discussed in the report.

A.1 Aluminum-Based Spent Nuclear Fuel

Aluminum-based spent nuclear fuel from research reactors will account for less than 1% of the total volume of SNF and high level waste that will require disposal in a geologic repository. However, much of the Al-SNF contains HEU with up to 93% enrichment. The Materials Test Reactor design assembly which is comprised of fuel elements or plates of aluminum-clad, aluminum-uranium alloy fuel is the dominant design (approximately 80% of total) and fuel material for research reactors. In addition, some reactor fuel assemblies were fabricated from aluminum-uranium silicide alloys or aluminum-uranium oxides. The fuel elements are clad with one of the aluminum alloys 1100, 5052, or 6061 or their foreign equivalents.

A.2 DOE SNF Repository Performance Categories

The DOE has categorized all of its spent fuel into fifteen categories. These categories were developed based upon fuel composition and characteristics [1]. The primary concern with grouping the spent fuels was assigning every fuel to a category and making certain that all of the spent fuel fit into a category. The total amount of DOE SNF, excepting Sodium bonded spent fuel (Category 14) and Navy spent fuel (Category 15), is approximately 2436 MTHM. This fuel (Categories 1-13) will be distributed among INEEL, Savannah River, and Hanford. Savannah River has responsibility for a total of 24.03 MTHM that includes fuel from Categories 5-7. Savannah River has 3.67 of the 87.93 MTHM comprising Category 5 and all of the fuels in Categories 6 and 7, 8.96 MTHM and 11.40 MTHM, respectively. The categorization of DOE SNF listed in Reference 1 is being re-categorized. The preliminary re-categorization lists Al-SNF as a single category. That is, Categories 5, 6, and 7 have been collapsed to one category based on their similar expected performance in the repository.

A.3 SRS Receipts—Al-SNF

A.3.1 Research Reactor Spent Nuclear Fuel Designs (Al-clad SNF Only)

Research reactors use a number of different fuel designs. These designs can be organized into three broad types: (1) materials and test reactor (MTR)-type design, which includes plate-type designs and concentric tube-type designs, (2) pin-type design, and (3) involute-type design. The following summarizes specific characteristics of the different types of fuel named above.

A.3.1.1 Plate-Type Design

This type of fuel design is used in the majority of research reactors. The thermal power of these reactors ranges from 1 MW to 50 MW. Figure A.1and Figure A.2 show typical fuel elements with this type of fuel design. The number of fuel plates in an element varies between 6 and 23, and the initial ^{235}U content varies between 37 g and 420 g per element. Similarly, the average burnup of a discharged spent nuclear fuel varies between 15 and 76 percent (^{235}U atom percent). The uranium enrichment used this type of fuel varies from just below 20 to 93 percent.

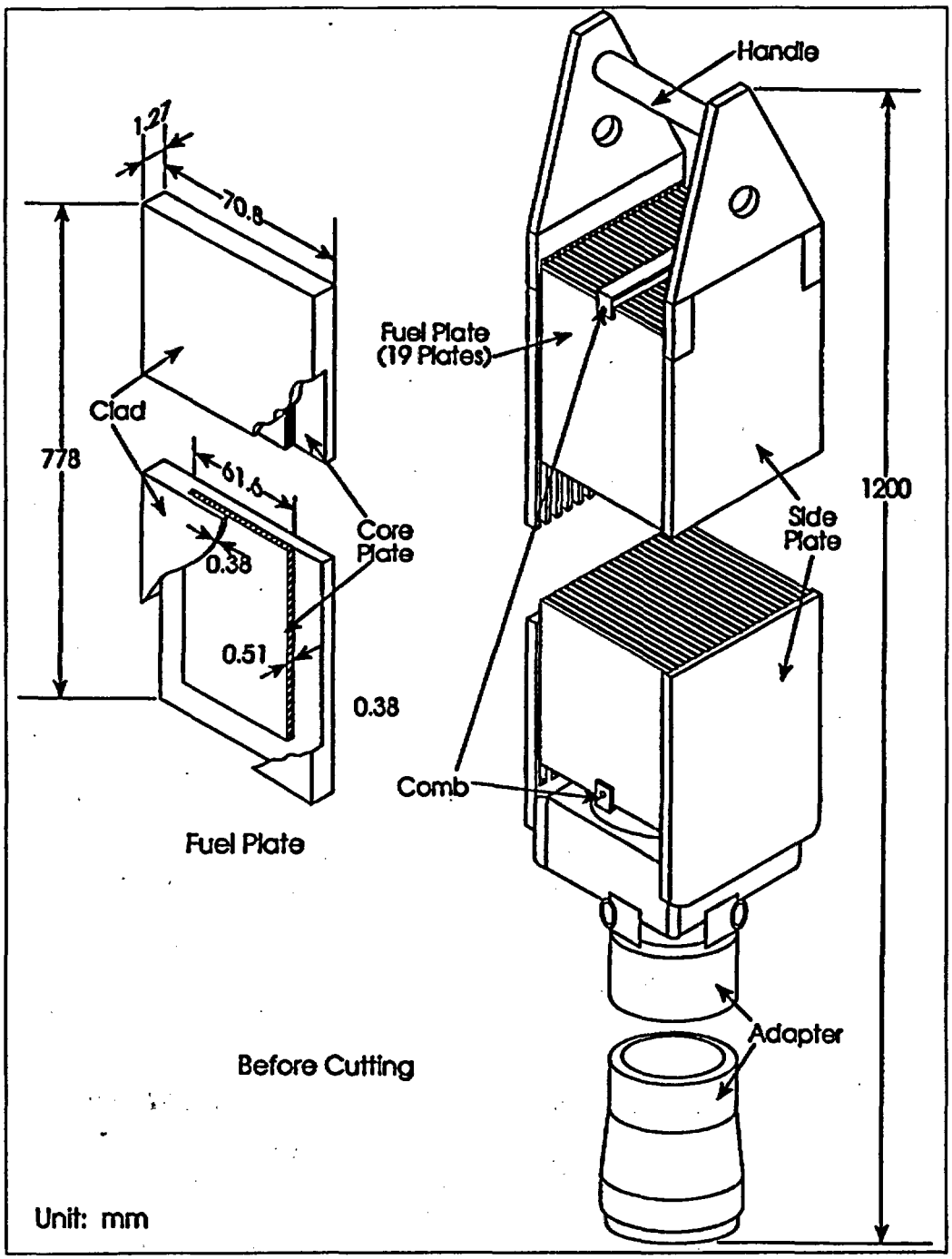
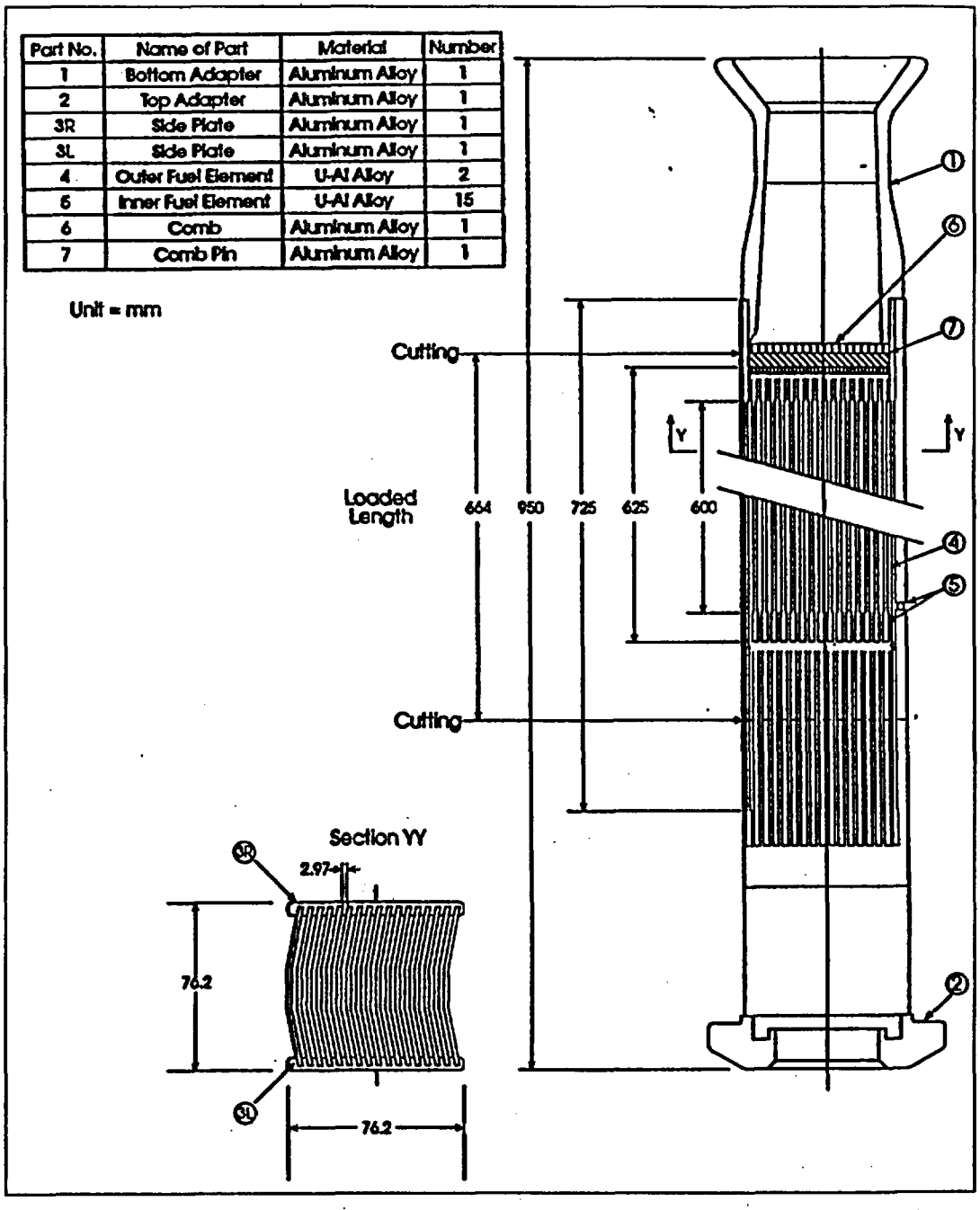


Figure A.1 Typical (Boxed-Type/Flat-Plate) Aluminum-Based Fuel Element Schematic



Part No.	Name of Part	Material	Number
1	Bottom Adapter	Aluminum Alloy	1
2	Top Adapter	Aluminum Alloy	1
3R	Side Plate	Aluminum Alloy	1
3L	Side Plate	Aluminum Alloy	1
4	Outer Fuel Element	U-Al Alloy	2
5	Inner Fuel Element	U-Al Alloy	15
6	Comb	Aluminum Alloy	1
7	Comb Pin	Aluminum Alloy	1

Figure A.2 Typical (Boxed-Type/Curved-Plate) Aluminum-Based Fuel Element Schematic

The following provides additional information on a typical plate-type spent nuclear fuel element that was used in a 50 MW research reactor, as shown in Figure A.1 and in Figure A.2.

The fuel element is made of an alloy of 23 percent by weight of 93 percent-enriched uranium in aluminum with a thin (0.38 mm) aluminum cladding. Each fuel element contains 19 fuel plates.

A.3.1.2 Concentric Tube Design

This type of fuel design is used in four foreign research reactors: Australian (HIFAR), Belgian (BR-2), Japanese (JRR-2) and Danish (DR-3).

The Belgian reactor is a 125 MW reactor, and the other three are each 10 MW. Figure A.3 shows a typical fuel element with concentric tube (tubular) fuel design type. The number of fuel tubes in an element varies between four (4) and six (6), and the initial ^{235}U content varies between 150 g and 400 g per element. The average burnup of discharged spent nuclear fuels from these reactors ranges between 47 and 55 percent (^{235}U atom percent). The uranium enrichment used in this fuel varies from just below 20 to 93 percent.

The following provides additional information on a typical tubular type spent nuclear fuel element (shown in Figure A.3 that was used in a 10 MW reactor).

This fuel element initially contains 220 g ^{235}U, and consists of five (5) concentric fuel tubes. Each tube is made of three curved fuel plates. The fuel is an alloy of uranium in aluminum with a thin (0.38 mm) aluminum cladding. Five (5) different curved fuel plate width sizes with 1.27 mm thickness and 625 mm height are used. The overall outside diameter of the outermost tube is 103 mm.

A.3.1.3 Pin-Type Design

Three types of foreign research reactors use pin-type design fuel. They are: the Canadian Safe LOW Power critical [K] Experiment (SLOWPOKE) (20 kW power); the Canadian NRU (125 MW power), NRX (24 MW power) and South Korean KMRR (30 MW) reactors. Among these reactors, the SLOWPOKE fuel pins are the smallest in size and uranium content.

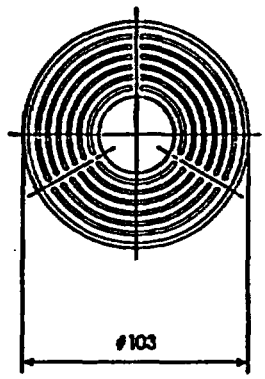
The SLOWPOKE reactor fuel pins have an outside diameter of 4.73 mm, a length of 220 mm, and contain 93 percent enriched uranium fuels. The ^{235}U content of each pin is 2.8 g. The maximum fuel burnup of discharged spent nuclear fuels is about 2 percent (^{235}U atom percent) in 10 to 20 years of reactor operation.

The SLOWPOKE spent nuclear fuel pins are usually bundled together in 10 to 15 pins per bundle. In the past, this fuel was shipped to the Savannah River Site in 50.8-mm outside diameter; 2.9-m long canisters containing between 150 to 160 pins per canister.

The fuel type in the Canadian research reactors consists of clusters of about 3 m long uranium aluminum alloy fuel pins clad in aluminum. The initial ^{235}U content of each fuel cluster varies between 491 g and 545 g. The current operating reactor (NRU) uses a fuel element that consists of a cluster of 12 long pins containing 491 g of ^{235}U per cluster. Each fuel pin has an overall length of 296 cm, and the fuel portion is 274.3 cm long. The fuel cluster, including the flow tube, is cut to a length of 292.6 cm before shipment. The average burnup of discharged spent nuclear fuels from a NRU reactor is about 76 percent (^{235}U atom percent). Figure A.4 shows a 12-pin cluster NRU fuel element. The fuel in the South Korean research reactor consists of two types of fuel clusters. The first is 18 pins per cluster with an initial ^{235}U content of 248 g (8.7 oz). The second is 36 pins per cluster with an initial ^{235}U content of 435 g (1 lb). The expected burnup of a discharged spent nuclear fuel from this reactor is approximately 65 percent (^{235}U atom percent).

Part No.	Name of Part	Material	Number
1	Bottom Adapter	Aluminum Alloy	1
2	Top Adapter	Aluminum Alloy	1
3	Outer Cylinder	Aluminum Alloy	1
4	Plate	Aluminum Alloy	3
5	Fuel Plate	———	5x3
6	Inner Layer Plate	Aluminum Alloy	3
7	Top Capsule Guide	Aluminum Alloy	1
8	Comb Plate	Aluminum Alloy	3
9	Capsule Guide	Aluminum Alloy	1
10	Comb	Aluminum Alloy	3

Unit = mm



Section AA

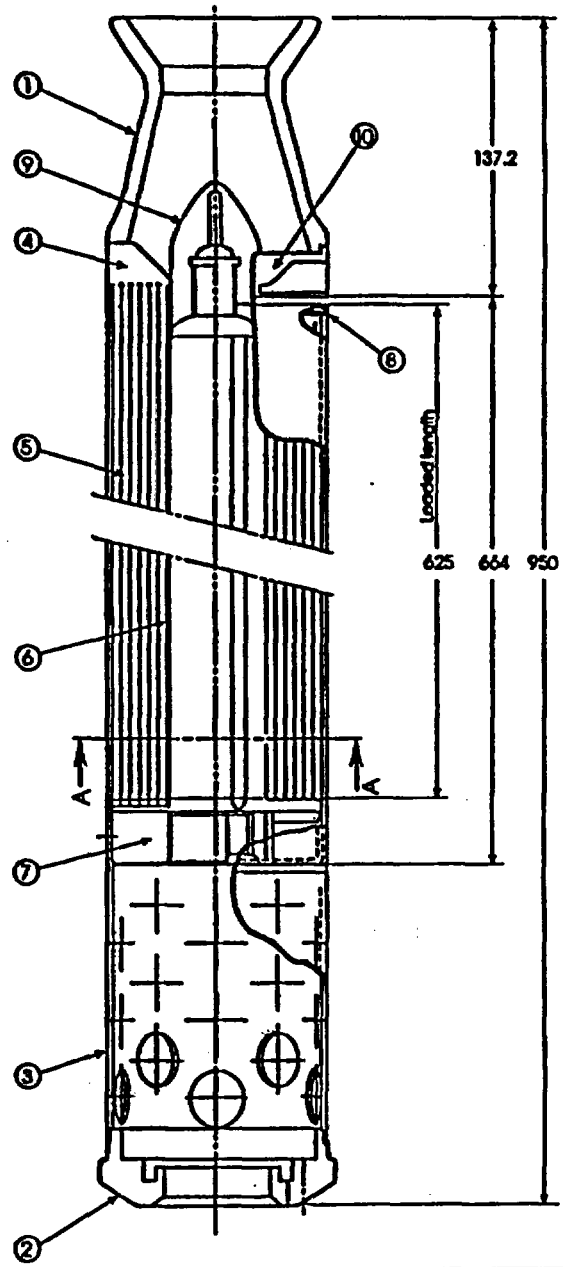


Figure A.3 Typical MTR (Tube-Type) Aluminum-Based Fuel Element Schematic

A.3.1.4 Involute Type Design

The fuel used in the high flux reactors is an involute-type fuel element. These research reactors consist of a single fuel element. There are currently two reactors of this design anticipated to ship SNF to the SRS. They are the Oak Ridge National Laboratory (ORNL) High Flux Isotope Reactor (100 MW) and the French Reactor à Haut Flux (57 MW).

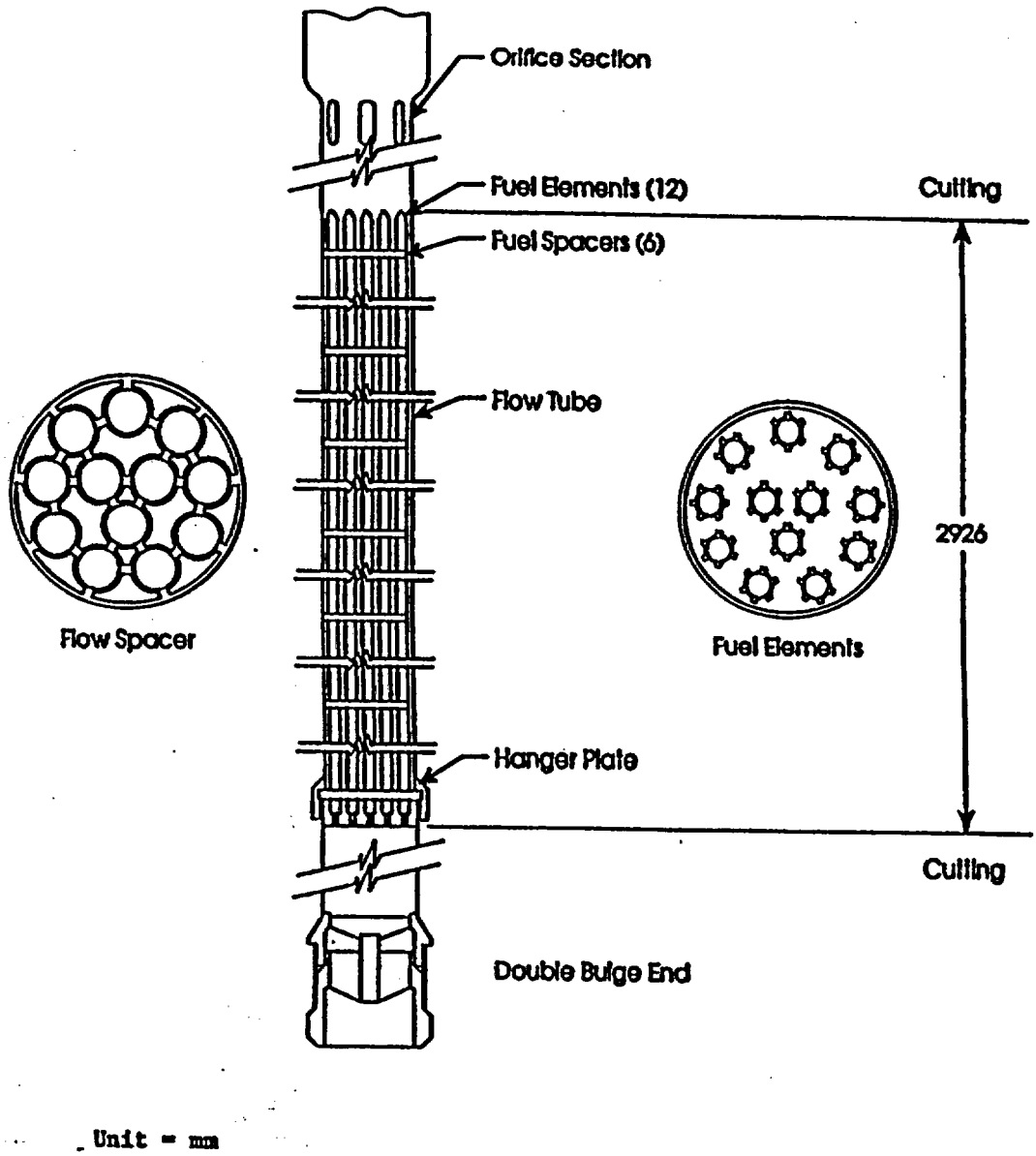


Figure A.4 Typical Pin-Type (Aluminum-Based) Fuel Element Schematic

The RHF fuel element contains 9.2 kg of uranium, enriched to 93 percent of ^{235}U in 280 involute fuel plates made of uranium aluminum alloy (UAl_3-Al), clad in aluminum. The weight of an element is about 100 kg. The fuel is in the annulus of two aluminum tubes: the inner tube has an outside diameter of 274 mm, and the outer tube has an outside diameter of 414 mm. The expected average burnup of a discharged spent nuclear fuel is 36 percent (^{235}U atom percent). Figure A.5 shows a schematic drawing of a configuration of annular fuel element similar to that of RHF fuels.

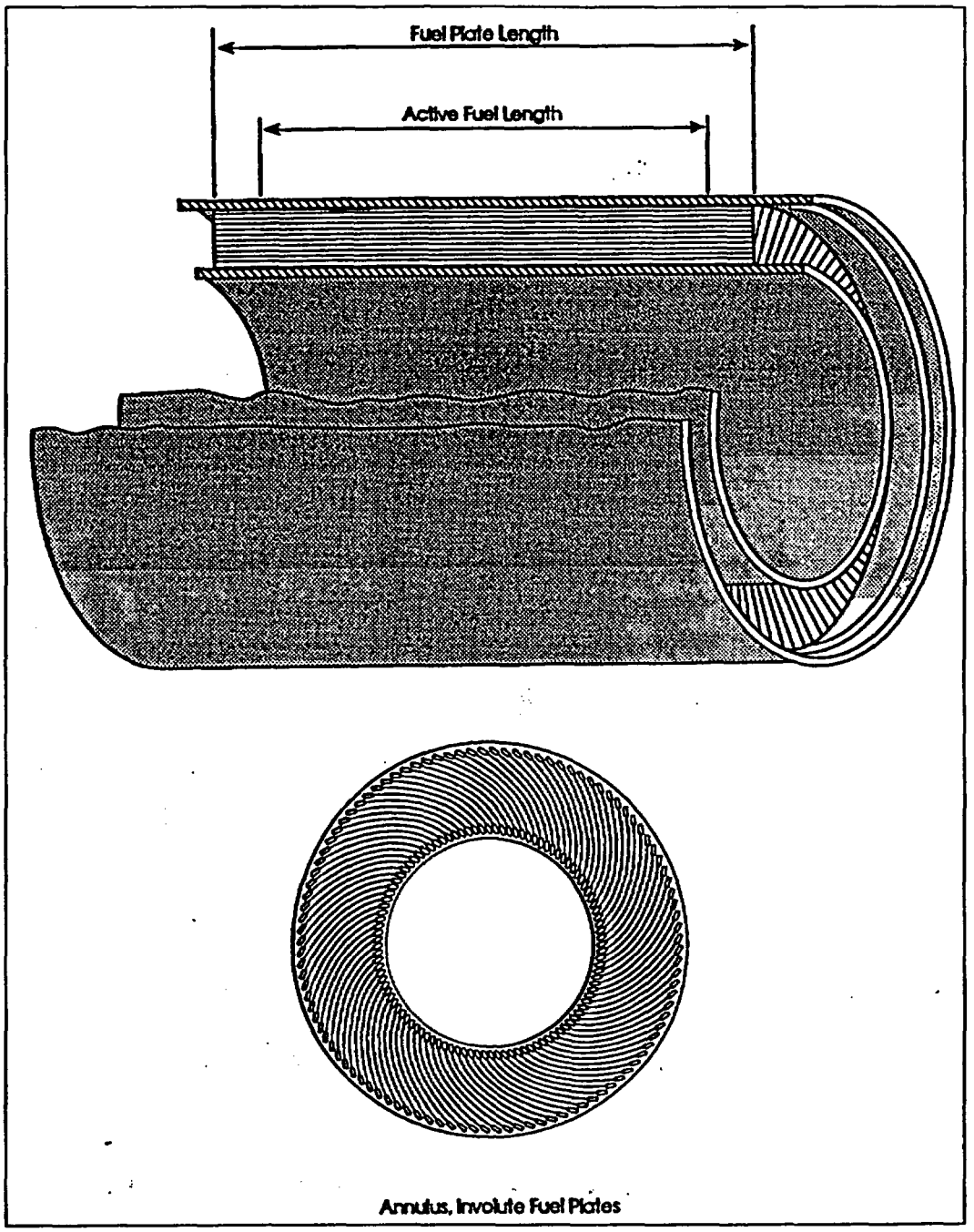


Figure A.5 Typical Involute(1)-Type (Aluminum-Based) Fuel Element Schematic

The HFIR fuel elements contain 10 kg of Uranium, enriched to 93% of ^{235}U. The element consists of an inner annulus (171 involute fuel plates) and an outer annulus (369 involute fuel plates). The material is uranium oxide - aluminum matrix, clad with aluminum. The total weight of the element is 136 kg. The tube is 80 cm long with an outer diameter of 43 cm. Figure A.6 shows a schematic drawing of a configuration of a typical HFIR fuel element.

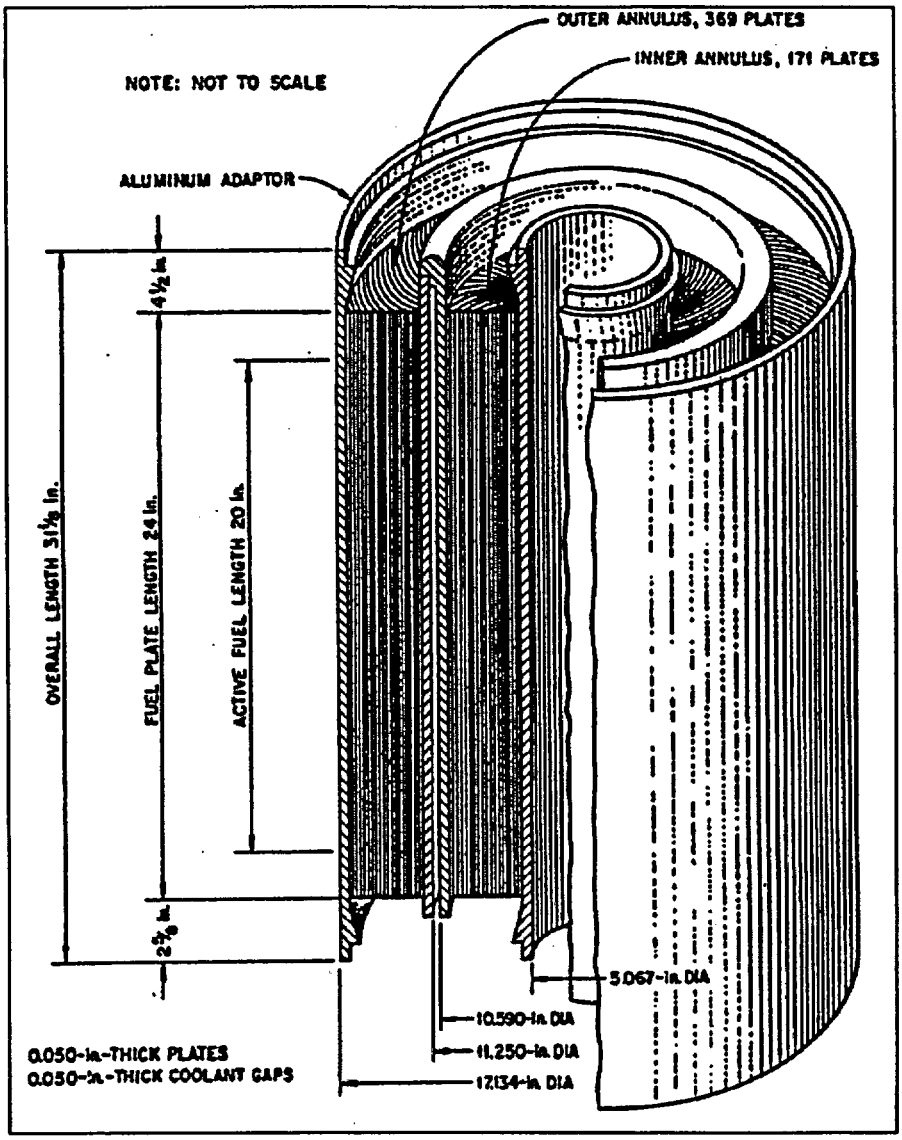


Figure A.6 Typical Involute(2)-Type (Aluminum-Based) Fuel Element Schematic

A.3.1.5 General Purpose Tubes

SNF elements may be received in SRS General Purpose Tubes, Square Cans, L-Basin Cans, or other aluminum bundling or canisterizing container. Onsite storage basins have utilized General Purpose Tubes to maximize storage space inventory, within radiological limits. The number of SNF elements within each General Purpose Tube may range from one to six, depending on the type and size of element(s). The SNF elements within a General Purpose Tube may be cropped and stacked with appropriate spacer material provided for radiological purposes. WSRC Drawings C-CS-L-0962, S4-2-609 and S5-2-6835 represent typical general-purpose tube design(s).

A.3.2 U Oxide/Failed Clad & Al

Table A.1, Table A.2, and Table A.3 provide a detailed listing of the current projection of types and quantities of uranium oxide, mixed oxide, and uranium metal fuel materials to be handled by TSF, respectively. The definition of the acronyms in the Reactor/Assembly Description column for these tables may be found in the International Atomic Energy Agency (IAEA) Directory of Nuclear Reactors and the IAEA Nuclear Research Reactors in the World. The tables may include some assemblies that are scheduled for reprocessing. The contents of the tables may change as processing is completed.

Table A.1 Uranium Oxide Inventory

Reactor/Assembly Description	Units	Configuration	Assembly Dimensions				Enrichment		U^{235} Mass (kg)		% Burnup	
			Hgt (cm)	Width (cm)	Lgth (cm)	Mass (kg)	% BOL ^{235}U	% EOL ^{235}U	BOL	EOL	U^{235}	Heavy Metal
					U_3O_8							
ASTRA (AUSTRIA)	14	MTR PLATE TYPE	8.05	7.61	83.00	6.00		19.95		0.35	0.0	0.0
BSR	41	19 PLATE MTR ASS'Y	8.05	7.61	85.41	4.48		85.71		0.15		
FRG-1 (GERMANY)	7	MTR TYPE		7.62	95.00	3.50	19.73	9.11	0.27	0.11	58.8	10.7
HFBR	220	18 CURVED PLATES	8.17	7.31	62.23	4.38	93.17	79.95	0.35	0.21	39.8	29.9
HFBR	20	18 CURVED PLATES	8.17	7.31	62.23	4.38	93.17	79.95	0.35	0.21	39.8	29.9
HFBR	350	18 CURVED PLATES	8.17	7.31	62.30	4.38	93.12	80.67	0.35	0.22	38.1	28.6
HFBR	2450	18 CURVED PLATES	8.17	7.31	62.30	4.38	93.09	93.09	0.35	0.35	0.0	0.0
HFBR	700	18 CURVED PLATES	8.17	7.31	62.30	4.38	93.12	80.67	0.35	0.22	38.1	28.6
HFIR	14	2 CONCENTRIC TUBES		43.50	80.00	139.90	93.10	86.55	9.40	6.84	27.2	21.7
HFIR	161	2 CONCENTRIC TUBES		43.50	80.00	139.90	92.95	92.95	9.39	6.58	30.0	30.0
'IST	126	17 CURVED PLATES	8.55	7.60	68.80	6.00	93.16	67.41	0.15	0.05	65.0	51.7
.ST	880	17 CURVED PLATES	8.55	7.60	68.80	6.00	93.33	61.96	0.35	0.10	71.1	56.5
OMEGA WEST (204)	16	18 OR 19 FLAT PLATES	8.39	7.66	108.00	4.90	93.14	86.38	0.19	0.14	28.6	22.7
OMEGA WEST (236)	44	18 OR 19 FLAT PLATES	8.39	7.66	108.00	4.90	93.22	82.85	0.22	0.14	37.8	30.0
OMEGA WEST (250)	27	18 OR 19 FLAT PLATES	8.39	7.66	108.00	4.90	93.14	87.38	0.23	0.17	27.7	23.0
ORR	17	19 CURVED PLATES	8.03	7.60	65.09	5.00	93.14	79.04	0.28	0.15	45.9	36.2
ORR	101	19 CURVED PLATES	8.03	7.60	65.09	5.00	90.70	81.19	0.28	0.17	38.5	31.3
ORR	100	19 CURVED PLATES	8.03	7.60	65.09	5.00		14.56		0.23		
ORR - MISC	10	SCRAP IN CANISTER				26.50		6.74		0.23		
RP-10 (PERU)	6	MTR-C		7.62	88.00	6.00	20.00	11.11	0.21	0.10	50.0	10.0
RP-10 (PERU)	23	MTR-S		7.62	88.00	6.00	20.00	11.11	0.23	0.14	39.1	10.0
RSG-GAS-30 (INDONESIA)	165	MTR-S		7.62	88.00	7.50	20.00	11.11	0.25	0.13	50.0	10.0
STERLING FOREST OXIDE	677	PARTICULATE	7.62	6.35	40.64	2.00		93.32		0.14		
UMRR	28	24 CURVED PLATES	7.57	8.74	87.00	6.20		88.38		0.15		
					UO_2							
FRR TARGET ARGENTINA	48	PARTICULATE				0.10	48.35	48.35	0.04	0.04	0.0	0.0
FRR TARGET CANADA	5952	PARTICULATE				0.10	48.35	48.35	0.04	0.04	0.0	0.0
FRR TARGET INDONESIA	48	PARTICULATE				0.10	48.35	48.35	0.04	0.04	0.0	0.0
HWCTR	45	TUBE		5.25	47.63	7.00		0.52		0.02		
WAPD (Na/K BONDED)	22	ROD		12.70	94.68	5.00		17.73		0.05		
TOTALS	12312					52970.24				2819.92		

Table A.2 Mixed Oxide Inventory

Reactor/Assembly Description	Units	Configuration	Fuel Material	Assembly Dimensions			Enrichment	U^{238} Mass (kg)	% Burnup	
				Width (cm)	Lgth (cm)	Mass (kg)	% EOL ^{235}U	EOL	U^{238}	Heavy Metal
EBR-II (MOX)	71	ROD	PUO2-UO2	0.59	154.94	2.00	78.74	0.03	4.7	2.9
SRE (U/TH)	37	CYLINDRICAL SLUGS	U-TH	8.89	280.04	103.00	74.27	4.00		
K/L/P NON-U TARGETS (PROCESSED)	104	ASSEMBLY	VARIOUS			23.83				
MARK 42 TARGETS (PU)	18	TUBE	PUO2	10.16	65.09	11.35				
TOTALS	230					6635.70		149.98		

Table A.3 Uranium Metal Inventory

Reactor/Assembly Description	Units	Configuration	Assembly Dimensions			Enrichment		U^{238} Mass (kg)		% Burnup	
			Width (cm)	Lgth (cm)	Mass (kg)	% BOL ^{235}U	% EOL ^{235}U	BOL	EOL	U^{238}	Heavy Metal
ARMF/CFRMF (CORE FILTER)	1	BLOCK	9.00	30.02	507.15		0.64		1.39		
EBR-II (DE-CLAD)	353	TUBE	1.59	398.48	75.00		0.22		0.10		
EBR-II TARGETS (PROCESSED)	1	ROD	12.40	421.60	562.79		0.25		0.48		
EBR-II TARGETS (PROCESSED)	1	ROD			562.79		0.25		0.71		
FRR TARGET (BELGIUM)	672	PARTICULATE			0.10	48.35	48.35	0.04	0.04	0.0	0.0
SRS TARGET SAMP. (PROCESSED)	42	UNKNOWN			9.50		0.18		0.02		
SRS TARGETS (PROCESSED)	6246	STACK-18 SLUG PAIRS			19.01	0.20	0.18		0.03		
SRS TARGETS (PROCESSED)	1224	STACK-18 SLUG PAIRS			19.01	0.20	0.19		0.03		
TRR (ROD 1) (PROCESSED)	81	ROD			108.00		0.63		0.34		
TRR (ROD 2) (PROCESSED)	310	ROD	12.70	304.80	220.00		0.63		0.33		
TOTALS	8931				2E+05				448.12		

A.3.3 UAl/Al

Table A.4 provides a detailed listing of the current projection of types and quantities of uranium aluminum fuel materials to be handled by TSF. The definition of the acronyms in the Reactor/Assembly Description column for this table may be found in the IAEA Directory of Nuclear Reactors and the IAEA Nuclear Research Reactors in the World. The table may include some assemblies that are scheduled for reprocessing. The contents of the table may change as processing is completed.

Table A.4 Uranium/Aluminum Inventory

Reactor/Assembly Description	Units	Configuration	Assembly Dimensions				Enrichment		U^{235} Mass (kg)		% Burnup	
			Hgt (cm)	Width (cm)	Lgth (cm)	Mass (kg)	% BOL ^{235}U	% EOL ^{235}U	BOL	EOL	U^{235}	Heavy Metal
ANLJ	19	ELEMENT		7.62	128.65	7.50	93.17	92.67	0.14	0.14	0.5	0.0
ARGONAUT REACTOR (BRAZIL)	28	MTR TYPE		7.62	95.00	5.50	20.00	19.98	0.20	0.20	0.1	0.0
ARMF	15	FLAT PLATES IN CAN	0.17	7.87	64.77	0.24	91.89	91.89	0.01	0.01	0.0	0.0
ARMF/CFRMF MARK I	56	15 FLAT PLATES	8.28	8.28	98.74	5.62	93.00	93.11		0.19		0.0
ARMF/CFRMF MARK I LL	2	15 FLAT PLATES	8.28	8.28	98.74	4.51	94.00	93.22		0.11		0.0
ARMF/CFRMF MARK II	8	15 FLAT PLATES	3.12	3.12	64.77	0.66	93.00	81.10		0.12		0.0
ARMF/CFRMF MARK III	4	15 FLAT PLATES	8.28	8.28	98.74	4.65	93.00	91.67		0.02		0.0
ASTRA (AUSTRIA)	14	MTR TYPE		7.62	95.00	4.40		90.00		0.31	67.0	62.3
ASTRA (AUSTRIA)	2	MTR TYPE		7.62	95.00	3.50		83.33		0.05	71.0	66.0
ASTRA (AUSTRIA)	22	MTR TYPE		7.62	95.00	4.90	93.00	82.71	0.28	0.10	64.0	59.5
ASTRA (AUSTRIA)	9	MTR TYPE		7.62	95.00	4.40		80.26		0.07	67.0	62.3
ASTRA (AUSTRIA)	5	MTR TYPE		7.62	95.00	5.30	45.00	22.26	0.32	0.11	65.0	29.2
ASTRA-(AUSTRIA)	26	19 FLAT PLATES	8.05	7.61	68.65	2.20	86.09	68.55	0.26	0.10	63.2	51.7
ATR	1096	19 CURVED PLATES	6.53	10.75	123.19	9.10	93.15	82.89	1.08	0.69	35.6	27.6
ATR	2780	19 CURVED PLATES	6.53	10.75	168.28	10.00	93.15	81.01	1.08	0.72	32.8	22.8
ATR	128	19 CURVED PLATES	6.53	10.75	123.19	9.10	93.15	79.96	1.08	0.62	42.4	32.9
ATR PLATES	30	MTR PLATE TYPE	0.25	10.75	121.90	2.00		92.00		0.01	0.0	0.0
ATSR	20	19 FLAT PLATES	8.20	7.61	64.50	2.45		93.18		0.15	0.2	0.0
BER-2 (GERMANY)	25	MTR-C TYPE		7.62	95.00	5.60	93.00	85.39	0.13	0.06	56.0	52.1
BER-2 (GERMANY)	71	MTR-S TYPE		7.62	95.00	5.50	93.00	85.39	0.18	0.08	56.0	52.1
BER-II (GERMANY)	66	MTR PLATE TYPE	8.10	7.62	90.00	5.40	93.12	75.90	0.17	0.08	51.3	40.2
BNL MEDICAL RX (BMRR)	68	18/19 CURVED PLATES	7.63	7.63	62.50	4.47		83.84		0.12		
BR-2 (BELGIUM)	16	ASSEMBLY		10.30	62.50	2.80	92.00	90.20	0.15	0.12	20.0	18.4
BR-2 (BELGIUM)	6	ASSEMBLY		10.30	62.50	4.10	90.00	90.00	0.14	0.14	0.0	0.0
BR-2 (BELGIUM)	5	ASSEMBLY		10.30	62.50	3.00	93.00	88.12	0.25	0.14	44.9	41.9
BR-2 (BELGIUM)	79	ASSEMBLY		10.30	62.50	3.60	93.00	88.04	0.34	0.22	36.0	33.1
BR-2 (BELGIUM)	1451	ASSEMBLY		10.30	62.50	3.80	93.00	87.36	0.40	0.21	48.0	44.6
BR-2 (BELGIUM)	40	ASSEMBLY		10.30	62.50	3.13	90.00	86.13	0.25	0.18	31.0	27.9
BR-2 (BELGIUM)	105	ASSEMBLY		10.30	62.50	5.80	93.00	84.52	0.39	0.21	46.0	41.9
BR-2 (BELGIUM)	64	ASSEMBLY		10.30	62.50	3.00	80.00	70.59	0.22	0.15	33.0	30.4
BSR	41	19 PLATE MTR ASS'Y	8.05	7.61	85.41	4.48		85.71		0.15		
DR-3 (DENMARK)	10	ASSEMBLY		10.30	62.50	2.70	90.00	86.92	0.15	0.07	50.0	46.5
DR-3 (DENMARK)	5	ASSEMBLY		10.30	62.50	2.50	90.00	86.68	0.12	0.06	51.0	47.4
DR-3 (DENMARK)	88	ASSEMBLY		9.34	62.50	2.80	89.10	69.38	0.15	0.07	53.0	39.8
DR-3 (DENMARK)	3	ASSEMBLY		9.34	62.50	2.80	19.88	10.28	0.18	0.08	54.6	12.1
ENEA (RANA-ITALY)	115	MTR TYPE	7.60	8.00	65.50	4.60		83.12		0.13	22.5	19.5
ENEA (RANA-ITALY)	33	MTR TYPE	7.61	8.04	65.50	4.90		19.42		0.13		
FMRB (GERMANY)	92	MTR TYPE		7.62	95.00	3.50	91.26	87.77	0.13	0.11	14.6	11.2
FRG-1 (GERMANY)	132	MTR TYPE		7.62	95.00	3.50	92.97	81.97	0.15	0.09	38.5	30.2
FRG-2 (GERMANY)	1	MTR-S TYPE		7.62	95.00	5.50	93.00	91.20	0.16	0.11	31.0	28.8
FRG-2 (GERMANY)	33	MTR-C TYPE		7.62	95.00	5.30	93.00	88.69	0.09	0.06	41.0	38.1
FRG-2 (GERMANY)	2	MTR-S TYPE		7.62	95.00	5.50	93.00	85.21	0.18	0.12	36.0	32.4

Reactor/Assembly Description	Units	Configuration	Assembly Dimensions				Enrichment		U^{235} Mass (kg)		% Burnup	
			Hgt (cm)	Width (cm)	Lgth (cm)	Mass (kg)	% BOL ^{235}U	% EOL ^{235}U	BOL	EOL	U^{238}	Heavy Metal
FRJ-2 (GERMANY)	200	ASSEMBLY		10.30	62.50	2.80	85.00	70.59	0.15	0.09	40.0	32.0
FRJ-2 (GERMANY)	200	ASSEMBLY		10.30	62.50	2.80	80.00	70.59	0.17	0.10	40.0	32.0
FRM (GERMANY)	10	MTR-S TYPE		7.62	95.00	5.50	93.00	91.20	0.16	0.11	31.0	28.8
FRM (GERMANY)	16	MTR-C TYPE		7.62	95.00	5.30	93.00	88.69	0.09	0.06	41.0	38.1
FRM (GERMANY)	13	MTR-S TYPE		7.62	95.00	5.60	93.00	85.10	0.23	0.10	57.0	53.0
FRM (GERMANY)	28	MTR-C TYPE		7.62	95.00	6.00	45.00	22.75	0.21	0.07	64.0	28.8
FRM (GERMANY)	74	MTR-S TYPE		7.62	95.00	5.80	45.00	22.75	0.28	0.10	64.0	28.8
FRR MTR ITALY	12	MTR TYPE		7.62	95.00	5.50		92.21		0.38		
FRR MTR ITALY	1	MTR TYPE		7.62	95.00	5.50		19.84		0.04		
FRR MTR SPAIN	1	MTR TYPE		7.62	95.00	5.50		49.63		0.13		
GENTR	16	STACKED DISKS		6.99	41.28	2.50	93.00	92.24	0.23	0.22	5.1	3.4
GRR-1 (GREECE)	107	MTR PLATE TYPE	7.62	7.62	77.80	4.90	91.96	84.60	0.16	0.11	28.3	22.0
GTRR	25	ASSEMBLY	7.04	7.52	69.85	10.74	93.07	90.27	0.19	0.16	14.1	11.4
HIFAR (AUSTRALIA)	187	ASSEMBLY		10.30	62.50	2.20	80.00	70.90	0.15	0.09	39.0	31.2
HIFAR (AUSTRALIA)	266	ASSEMBLY		10.30	62.50	2.20	90.00	65.83	0.12	0.05	53.0	42.4
HIFAR (AUSTRALIA)	52	ASSEMBLY		10.30	62.50	2.30	60.00	43.30	0.15	0.08	49.0	29.4
HIFAR (AUSTRALIA)	169	ASSEMBLY		10.30	62.50	2.30	60.00	41.35	0.17	0.08	53.0	31.8
HIFAR (AUSTRALIA)	289	ASSEMBLY		10.30	62.50	2.30	20.00	11.11	0.20	0.10	50.0	10.0
HFR-PETTEN (NETHERLANDS)	715	MTR-S TYPE		7.62	95.00	4.60	93.00	83.21	0.42	0.21	51.0	46.4
HFR-PETTEN (NETHERLANDS)	161	MTR-C TYPE		7.62	95.00	4.50	93.00	80.18	0.29	0.12	60.0	54.6
HOR (NETHERLANDS)	61	MTR-S TYPE		7.62	95.00	4.30	93.00	86.92	0.19	0.09	50.0	46.5
HOR (NETHERLANDS)	19	MTR-S TYPE		7.62	95.00	4.50	90.00	86.92	0.10	0.05	50.0	46.5
HOR (NETHERLANDS)	33	ASSEMBLY	7.62	7.62	88.00	5.50	93.13	77.50	0.18	0.09	49.0	38.8
IAE-R1 (BRAZIL)	4	MTR-O TYPE	7.62	7.62	95.00	6.50	93.00	91.42	0.13	0.10	20.0	18.6
IAE-R1 (BRAZIL)	6	MTR-C TYPE	7.62	7.62	95.00	6.50	93.00	90.33	0.09	0.06	30.0	27.9
IAE-R1 (BRAZIL)	33	MTR-S TYPE		7.62	95.00	5.60	93.00	90.29	0.18	0.13	30.0	27.9
IAE-R1 (BRAZIL)	6	MTR-C TYPE		7.62	95.00	6.50	20.00	19.99	0.08	0.08	0.0	0.0
IAE-R1 (BRAZIL)	33	MTR-S1 TYPE		7.62	95.00	5.60	20.00	16.67	0.16	0.13	20.0	4.0
IAE-R1 (BRAZIL)	34	MTR-S1 TYPE		7.62	95.00	5.60	20.00	16.32	0.16	0.12	22.0	4.4
IAE-R1 (BRAZIL)	5	MTR-S2 TYPE		7.62	95.00	5.60	20.00	15.43	0.18	0.13	27.0	5.4
IAE-R1 (BRAZIL)	6	MTR-C TYPE		7.62	95.00	6.50	20.00	13.23	0.08	0.05	39.0	7.8
IAN-R1 (COLUMBIA)	21	MTR TYPE		7.62	95.00	3.40		90.55		0.13	0.7	0.6
IRR-1 (ISRAEL)	31	MTR-C TYPE		7.62	95.00	4.80	93.00	84.16	0.16	0.06	60.0	55.8
IRR-1 (ISRAEL)	122	MTR-S TYPE		7.62	88.00	4.80	93.00	84.16	0.22	0.09	60.0	55.8
ISU - ARGONAUT	22	FLAT PLATES IN CAN		8.26	65.30	8.00	93.34	93.34	0.15	0.15	0.0	0.0
ISU - ARGONAUT	22	CAN		8.26	66.28	5.00	93.00	93.32		0.28	0.0	0.0
JEN-1 (SPAIN)	32	MTR TYPE	7.73	7.73	103.00	6.00	42.72	40.00	0.14	0.13	6.4	2.3
JEN-1 (SPAIN)	8	MTR TYPE	7.73	7.73	103.00	6.00	17.93	14.81	0.13	0.11	17.4	2.9
JMTR (JAPAN)	9	MTR-S TYPE		7.62	95.00	5.50	93.00	91.30	0.28	0.23	21.0	19.5
JMTR (JAPAN)	131	10 FLAT PLATES	7.62	7.62	120.00	6.00	93.21	88.51	0.27	0.21	21.2	17.0
JMTR (JAPAN)	165	MTR-C TYPE		7.62	95.00	4.50	45.00	39.86	0.21	0.17	19.0	8.6
JMTR (JAPAN)	675	MTR-S TYPE		7.62	95.00	6.00	45.00	37.39	0.32	0.24	27.0	12.1
JMTRC (JAPAN)	30	MTR TYPE		7.62	95.00	6.00	45.00	44.98	0.32	0.32	0.1	0.0
JRR-2 (JAPAN)	28	MTR TYPE		7.62	95.00	5.00	93.00	87.56	0.19	0.10	47.0	43.7
JRR-2 (JAPAN)	7	CONCENTRIC TUBES			66.40	5.00	93.00	87.56	0.19	0.10	47.0	43.7

Reactor/Assembly Description	Units	Configuration	Assembly Dimensions				Enrichment		U^{235} Mass (kg)		% Burnup	
			Hgt (cm)	Width (cm)	Lgth (cm)	Mass (kg)	% BOL ^{235}U	% EOL ^{235}U	BOL	EOL	U^{235}	Heavy Metal
JRR-2 (JAPAN)	138	CONCENTRIC TUBES			66.40	6.00	45.00	30.25	0.22	0.12	47.0	21.1
JRR-3(M) (JAPAN)	99	MTR-C TYPE		7.62	95.00	5.00	20.00	11.11	0.19	0.09	50.0	10.0
JRR-3(M) (JAPAN)	506	MTR-S TYPE		7.62	95.00	8.00	20.00	11.11	0.30	0.15	50.0	10.0
JRR-4 (JAPAN)	54	MTR TYPE		7.62	95.00	5.00	93.00	91.40	0.17	0.13	20.0	18.6
KUR (JAPAN)	54	MTR-C TYPE		7.62	95.00	4.30	93.00	91.24	0.09	0.07	22.0	20.5
KUR (JAPAN)	218	MTR-S TYPE		7.62	95.00	4.60	93.00	90.98	0.18	0.14	24.0	22.3
LFR (NETHERLANDS)	14	MTR TYPE		7.62	95.00	5.00	93.00	92.99	0.21	0.21	0.1	0.1
MINERVE (FRANCE)	32	MTR TYPE		7.62	95.00	5.00	93.00	92.99	0.25	0.25	0.0	0.0
MIT	277	15 FLAT PLATES	6.42	6.11	66.68	4.00	93.00	86.69	0.51	0.40	22.1	16.5
MIT	72	15 FLAT PLATES	6.11	6.11	66.68	4.00	93.39	83.73	0.47	0.29	39.2	32.2
MNR (CANADA)	23	MTR-C TYPE		7.62	95.00	3.67	90.00	89.17	0.11	0.07	38.0	35.3
MNR (CANADA)	53	MTR-S TYPE		7.62	95.00	3.92	93.00	89.17	0.20	0.12	38.0	35.3
MNR (CANADA)	21	MTR-S TYPE		7.62	95.00	3.92	93.00	89.17	0.20	0.12	38.0	35.3
MNR (CANADA)	41	MTR TYPE		7.62	95.00	3.92	93.07	81.35	0.18	0.11	39.7	31.0
MOATA ARGONAUT (AUSTRALIA)	12	ASSEMBLY		7.62	88.00	4.30	90.00	89.99	0.25	0.25	0.1	0.1
MRU/WMA (CANADA)	741	MULTI-PIN CLUSTER				4.70	93.00	71.62	0.50	0.09	81.0	75.3
MURR	32	24 CURVED PLATES	7.75	9.14	82.55	6.38	93.50	87.47	0.73	0.60	18.3	12.9
MURR	24	24 CURVED PLATES	7.75	9.14	82.55	6.38	93.50	87.16	0.73	0.59	20.1	14.3
MURR	184	24 CURVED PLATES	7.47	8.74	87.00	6.20	93.14	83.50	0.77	0.60	22.6	17.7
MURR (MTR-SI)	792	24 CURVED PLATES	7.04	14.63	82.55	6.20	93.00	90.15	0.77	0.58	25.1	22.8
NRCRR (IRAN)	7	MTR-C TYPE		7.62	95.00	4.10	94.50	90.29	0.11	0.08	30.0	27.9
NRCRR (IRAN)	22	MTR-S TYPE		7.62	95.00	4.10	93.00	87.00	0.20	0.14	30.0	27.9
NRX (CANADA)	131	MULTI-PIN CLUSTER				4.50	93.00	83.78	0.55	0.21	62.0	57.8
OHIO STATE	24	18 FLAT PLATES		7.62	88.90	50.00	93.25	93.24	0.13	0.13	0.1	0.0
ORPHEE (FRANCE)	148	MTR-C TYPE		7.62	95.00	8.00	93.00	90.39	0.63	0.44	30.0	27.9
ORPHEE (FRANCE)	148	MTR-S TYPE		7.62	95.00	9.00	93.00	90.39	0.84	0.59	30.0	27.9
PARR (PAKISTAN)	17	MTR-C TYPE		7.62	95.00	4.10	92.00	88.20	0.11	0.07	35.0	32.2
PARR (PAKISTAN)	65	MTR-S TYPE		7.62	95.00	4.10	92.00	88.20	0.20	0.13	35.0	32.2
PRR-1 (PHILIPPIINES)	20	MTR TYPE		7.62	95.00	4.10	93.00	90.10	0.16	0.12	25.0	23.2
PRR-1 (PHILIPPIINES)	30	MTR TYPE		7.62	95.00	4.50	20.00	16.68	0.14	0.11	20.0	4.0
PTR (CANADA)	14	MULTI-PIN CLUSTER				3.80	93.00	93.00	0.15	0.15	0.5	0.5
PTR (CANADA)	225	MULTI-PIN CLUSTER				4.70	93.00	76.54	0.49	0.12	76.0	70.8
PURDUE UNIVERSITY	124	10 FLAT PLATES	0.15	7.01	68.80	0.03	93.00	93.22	0.17	0.02	90.0	90.0
R-2 (SWEDEN)	6	MTR-C TYPE		7.62	95.00	4.30	90.00	88.44	0.12	0.10	15.0	13.5
R-2 (SWEDEN)	21	MTR-S TYPE		7.62	95.00	4.30	90.00	88.44	0.12	0.10	15.0	13.5
R-2 (SWEDEN)	48	MTR-C TYPE		7.62	95.00	4.60	93.00	76.12	0.16	0.04	76.0	70.7
R-2 (SWEDEN)	304	MTR TYPE	7.62	7.62	92.40	6.00	93.06	74.77	0.24	0.10	58.8	48.8
RA-3 (ARGENTINA)	36	MTR-C TYPE		7.62	95.00	4.30	90.00	85.40	0.15	0.10	35.0	31.5
RA-3 (ARGENTINA)	144	MTR-S TYPE		7.62	95.00	4.30	90.00	85.40	0.19	0.13	35.0	31.5
RA-6 (ARGENTINA)	14	MTR-C TYPE		7.62	95.00	4.30	90.00	86.31	0.15	0.11	30.0	27.0
RA-6 (ARGENTINA)	59	MTR-S TYPE		7.62	95.00	4.30	90.00	86.31	0.19	0.14	30.0	27.0
RA-O CORDOBA UNIV (ARGENTINA)	30	MTR TYPE		7.62	95.00	4.30	20.00	19.84	0.13	0.12	1.0	0.2
RECH-1 (CHILE)	28	MTR TYPE	7.61	7.61	99.30	5.00	80.00	61.00	0.16	0.08	48.6	32.6
RHF (FRANCE)	86	ASSEMBLY		41.40		110.00	90.00	88.85	8.50	5.10	40.0	37.2
RHF (FRANCE)	4	2 CONCENTRIC TUBES		40.64	97.00	102.00	92.97	81.44	8.58	5.19	39.4	30.9

Reactor/Assembly Description	Units	Configuration	Assembly Dimensions				Enrichment		U^{238} Mass (kg)		% Burnup	
			Hgt (cm)	Width (cm)	Lgth (cm)	Mass (kg)	% BOL ^{235}U	% EOL ^{235}U	BOL	EOL	U^{235}	Heavy Metal
RINSC	66	18 FLAT PLATES		7.62	62.50	2.00	93.14	90.55	0.12	0.11	11.6	9.1
RP-1 (PORTUGAL)	3	MTR-O TYPE		7.62	95.00	4.60	20.00	17.78	0.09	0.08	11.1	0.0
RP-1 (PORTUGAL)	9	MTR-C TYPE		7.62	95.00	5.30	20.00	17.18	0.09	0.07	17.0	3.4
RP-1 (PORTUGAL)	45	MTR-S TYPE		7.62	95.00	4.90	20.00	16.14	0.18	0.14	23.0	4.6
RPI (PORTUGAL)	9	MTR-C TYPE		7.62	95.00	4.90	93.00	88.85	0.15	0.09	40.0	37.2
RPI (PORTUGAL)	22	MTR-S TYPE		7.62	95.00	4.50	93.00	88.85	0.26	0.16	40.0	37.2
RU-1 (URAGUAY)	19	MTR TYPE		7.62	95.00	5.00	20.00	20.00	0.19	0.19	0.0	0.0
RV-1 (VENEZUELA)	120	MTR TYPE		7.62	95.00	6.00	20.00	16.66	0.14	0.11	20.0	4.0
SAFARI-1 (SO. AFRICA)	4	MTR-O TYPE	7.62	7.62	95.00	5.10	90.00	89.10	0.09	0.05	38.5	35.8
SAFARI-1 (SO. AFRICA)	40	MTR-S TYPE		7.62	95.00	4.20	90.00	86.78	0.20	0.10	50.6	47.1
SAFARI-1 (SO. AFRICA)	6	MTR-C TYPE		7.62	95.00	4.50	90.00	82.71	0.14	0.05	64.0	59.5
SAPHIR (SWITZERLAND)	128	MTR PLATE TYPE	8.05	7.62	87.50	4.11	62.08	35.13	0.28	0.11	61.9	32.6
SAPHIR (SWITZERLAND)	7	MTR-C1 TYPE		7.62	95.00	4.00	93.00	82.82	0.17	0.06	64.0	59.5
SAPHIR (SWITZERLAND)	11	MTR-C2 TYPE		7.62	95.00	4.20	93.00	82.71	0.21	0.07	64.0	59.5
SAPHIR (SWITZERLAND)	55	MTR-S TYPE		7.62	95.00	3.90	93.00	80.46	0.28	0.09	69.0	64.2
SCARABEE (FRANCE)	31	MTR TYPE		7.62	95.00	5.50	93.00	92.93	0.31	0.30	1.0	0.9
SILOE/MELUSINE (FRANCE)	97	MTR-C TYPE		7.62	95.00	5.40	93.00	87.96	0.25	0.14	45.0	41.9
SILOETTE (FRANCE)	30	MTR TYPE		7.62	95.00	5.90	93.00	92.97	0.33	0.33	0.5	0.5
SLOWPOKE-ALBERTA (CANADA)	2	ASSEMBLY				2.30	93.00	93.00	0.41	0.41	1.5	1.4
SLOWPOKE-HALIFAX (MONTREAL)	2	ASSEMBLY				2.40	93.00	93.00	0.44	0.43	1.5	1.4
SLOWPOKE-JAMAICA (JAMAICA)	2	ASSEMBLY				2.30	93.00	93.00	0.41	0.40	1.5	1.4
SLOWPOKE-KANATA AT AECL (CANADA)	2	ASSEMBLY				2.40	93.00	93.00	0.44	0.43	1.5	1.4
SLOWPOKE-MONTREAL (CANADA)	2	ASSEMBLY				2.40	93.00	93.00	0.44	0.43	1.5	1.4
SLOWPOKE-SASKATCHWAN (CANADA)	2	ASSEMBLY				2.40	93.00	93.00	0.44	0.43	1.5	1.4
SLOWPOKE-TORONTO (CANADA)	2	ASSEMBLY				2.10	93.00	93.00	0.39	0.38	1.5	1.4
SRS (PROCESSED)	5	TUBE				10.00	32.00	31.70		0.39		6.2
SRS DRIVER FUEL	449	ASSEMBLY				23.83		66.22		2.05		
SRS DRIVER FUEL	516	SCRAP				33.43		59.62		3.00		
SRS DRIVER FUEL	4	ASSEMBLY				1.32		31.05		0.06		
SRS DRIVER FUEL (PROCESSED)	902	ASSEMBLY				22.62		65.29		2.34		
SRS DRIVER FUEL (PROCESSED)	13	ASSEMBLY				33.08		61.18		3.18		
STERLING FOREST	200	ASSEMBLY				2.88		84.01		0.12		
STRASBOURG-CRONENBOURG (FRANCE)	48	MTR TYPE		7.62	95.00	5.00	90.00	89.98	0.12	0.12	0.5	0.5
THAR-ARGONAUT (TAIWAN)	23	MTR TYPE		7.62	95.00	3.00	20.00	19.83	0.30	0.30	0.0	0.0
THOR (TAIWAN)	34	MTR TYPE		7.62	95.00	5.10	93.00	90.77	0.14	0.10	26.0	24.2
TR-2 (TURKEY)	8	MTR-C TYPE		7.62	95.00	4.80	93.00	86.92	0.21	0.10	50.0	46.5
TR-2 (TURKEY)	2	MTR-O TYPE		7.62	95.00	4.80	93.00	86.92	0.17	0.09	50.0	46.5
TR-2 (TURKEY)	18	MTR-S TYPE		7.62	95.00	4.80	93.00	86.92	0.28	0.14	50.0	46.5
TRR-1/M-1 (THAILAND)	31	MTR TYPE		7.62	95.00	4.20	90.00	87.53	0.15	0.12	22.0	19.8

Reactor/Assembly Description	Units	Configuration	Assembly Dimensions				Enrichment		U^{235} Mass (kg)		% Burnup	
			Hgt (cm)	Width (cm)	Lgth (cm)	Mass (kg)	% BOL ^{235}U	% EOL ^{235}U	BOL	EOL	U^{235}	Heavy Metal
TSR FUEL	1	SPHERICAL		73.66		182.00		93.48		8.60		
TTR-1 (JAPAN)	27	MTR TYPE		7.62	95.00	6.50	20.00	19.94	0.13	0.13	0.4	0.1
ULYSSE-ARGONAUT (FRANCE)	48	MTR TYPE		7.62	95.00	5.00	90.00	89.98	0.12	0.12	0.5	0.5
UNIV OF FLORIDA (ARGONAUT)	256	11 FLAT PLATES	5.44	7.23	65.09	1.22	93.00	93.10	0.02	0.01	9.9	10.0
UNIV OF MASS-LOWELL	26	18 FLAT PLATES	7.62	7.62	101.60	8.85		92.71		0.15		
UNIV OF MICHIGAN	156	18 CURVED PLATES	7.47	8.26	87.38	4.56	19.75	15.26	0.17	0.11	33.6	14.1
UNIV OF MICHIGAN	48	18 CURVED PLATES	7.47	8.26	87.38	6.00	19.85	14.67	0.15	0.10	30.0	5.3
UNIV OF VIRGINIA	44	22 FLAT PLATES	8.26	7.93	95.38	6.50	92.94	86.92	0.17	0.14	17.6	13.5
UNIV OF WASHINGTON	26	11 FLAT PLATES	5.98	7.24	68.90	2.74		93.07		0.14		
UTR 300 (UNIV. OF GLASGOW -UK)	12	MTR TYPE		7.62	95.00	6.50	90.00	89.82	0.30	0.29	2.0	1.8
UTR-10 KINKI (JAPAN)	12	MTR TYPE	12.30	7.62	64.80	4.20	93.00	93.00	0.27	0.27	0.0	0.0
WORCESTER POLY INSTITUTE	26	18 CURVED PLATES	7.75	7.75	101.60	5.80	19.90	19.79		0.17		2.2
ZPRL (TAIWAN)	35	MTR TYPE		7.62	95.00	5.15	93.20	93.19	0.13	0.13	0.1	0.1
ZPRL (TAIWAN)	35	MTR TYPE		7.62	95.00	5.15	20.00	19.95	0.13	0.12	0.3	0.1
TOTALS	19763					162796.28		72.42		10505.90		

A.3.4 U-Si

Table A.5 provides a detailed listing of the current projection of types and quantities of uranium silicon fuel materials to be handled by TSF, respectively. The definition of the acronyms in the Reactor/Assembly Description column for this table may be found in the IAEA Directory of Nuclear Reactors and the IAEA Nuclear Research Reactors in the World. The table may include some assemblies that are scheduled for reprocessing. The contents of the table may change as processing is completed.

Table A.5 Uranium/Silicon Inventory

Reactor/Assembly Description	Units	Configuration	Assembly Dimensions				Enrichment		U^{235} Mass (kg)		% Burnup	
			Hgt (cm)	Width (cm)	Lgth (cm)	Mass (kg)	% BOL ^{235}U	% EOL ^{235}U	BOL	EOL	U^{235}	Heavy Metal
ASTRA (AUSTRIA)	5	MTR TYPE	8.02	7.61	87.30	6.00		8.00		0.07	60.0	12.0
ASTRA (AUSTRIA)	15	MTR TYPE	8.05	7.61	87.30	6.40		8.00		0.09	60.0	12.0
ASTRA (AUSTRIA)	71	MTR TYPE	8.05	7.61	87.30	6.40		8.00		0.12	60.0	12.0
DR-3 (DENMARK)	556	TUBES		10.30	62.50	3.30	20.00	9.71	0.18	0.08	57.0	11.4
DR-3 (DENMARK)	135	TUBES		10.30	62.50	4.00	20.00	11.11	0.18	0.09	50.0	10.0
FRG-1 (GERMANY)	96	MTR-S TYPE	7.62	7.62	88.00	6.70	20.00	7.83	0.32	0.11	66.0	13.2
FRG-1 (GERMANY)	18	MTR TYPE		7.62	95.00	3.50	19.83	9.54	0.28	0.12	56.9	10.4
FRJ-2 (GERMANY)	18	TUBES		10.30	62.50	4.00	20.00	11.11	0.20	0.10	50.0	10.0
FRJ-2 (GERMANY)	135	TUBES		10.30	62.50	4.00	20.00	11.11	0.22	0.11	50.0	10.0
GRR-1 (GREECE)	18	MTR-C TYPE	7.62	7.62	88.00	5.70	20.00	9.09	0.12	0.08	35.0	7.0
GRR-1 (GREECE)	62	MTR-S TYPE	7.62	7.62	88.00	5.70	20.00	9.09	0.22	0.14	35.0	7.0
HFR-PETTEN (NETHERLANDS)	72	MTR-C TYPE		7.62	88.00	4.50	20.00	9.09	0.31	0.12	60.0	12.0

Reactor/Assembly Description	Units	Configuration	Assembly Dimensions				Enrichment		U^{235} Mass (kg)		% Burnup	
			Hgt (cm)	Width (cm)	Lgth (cm)	Mass (kg)	% BOL ^{235}U	% EOL ^{235}U	BOL	EOL	U^{235}	Heavy Metal
HFR-PETTEN (NETHERLANDS)	363	MTR-S2 TYPE	7.62	7.62	88.00	4.60	20.00	9.09	0.45	0.22	51.0	10.2
HOR (NETHERLANDS)	7	MTR-C TYPE		7.62	88.00	5.10	20.00	9.09	0.16	0.06	60.0	12.0
HOR (NETHERLANDS)	43	MTR-S TYPE		7.62	88.00	5.50	20.00	9.09	0.30	0.12	60.0	12.0
IR-1 (ISRAEL)	8	MTR-C TYPE	7.62	7.62	88.00	4.80	20.00	9.09	0.31	0.11	65.0	13.0
ISU - ARGONAUT	13	17 FLAT PLATES	14.06	7.62	66.04	2.33		19.77		0.31	2.5	2.4
JMTR (JAPAN)	149	10 FLAT PLATES		7.62	88.00	5.40	20.00	14.89	0.28	0.19	30.0	6.0
JMTR (JAPAN)	574	ASSEMBLY		7.62	88.00	7.30	20.00	14.89	0.41	0.29	30.0	6.0
JRR-4 (JAPAN)	47	MTR TYPE		7.62	88.00	6.00	20.00	15.79	0.20	0.15	25.0	5.0
KUR (JAPAN)	17	MTR-C TYPE	7.62	7.62	88.00	4.60	20.00	16.83	0.10	0.09	19.0	3.8
KUR (JAPAN)	70	MTR-S TYPE		7.62	88.00	5.40	20.00	16.49	0.21	0.17	21.0	4.2
MNR (CANADA)	8	MTR-C TYPE		7.62	88.00	4.90	20.00	9.09	0.16	0.08	50.0	10.0
MNR (CANADA)	35	MTR-S TYPE		7.62	88.00	5.40	20.00	9.09	0.29	0.14	50.0	10.0
NEREIDE (FRANCE)	46	12 CURVED PLATES	7.98	7.57	87.30	62.50	19.85	19.80	0.15	0.15	0.3	0.1
NRU (CANADA)	1527	MULTI-PIN CLUSTER				6.60	20.00	7.83	0.49	0.12	76.0	15.0
OHIO STATE	414	18 FLAT PLATES		7.62	88.90	0.48	19.75	19.73	0.20	0.01	93.8	93.8
ORR	32	19 CURVED PLATES	8.03	7.60	65.09	5.00	19.85	19.82	0.32	0.32	0.0	0.0
ORR - MISC	10	SCRAP IN CANISTER				26.50		6.74		0.23		
OSIRIS (FRANCE)	177	MTR-C TYPE	7.62	7.62	88.00	7.00	20.00	9.09	0.39	0.17	55.0	11.0
OSIRIS (FRANCE)	724	MTR-S TYPE	7.62	7.62	88.00	7.50	20.00	9.09	0.51	0.23	55.0	11.0
PURDUE UNIVERSITY	22	10 FLAT PLATES	0.15	7.01	68.80	1.70	19.00	19.00	0.22	0.22	0.0	0.0
R-2 (SWEDEN)	580	MTR-C TYPE	7.62	7.62	88.00	6.20	20.00	9.09	0.40	0.16	60.0	12.0
R-2 (SWEDEN)	82	MTR-C TYPE	7.62	7.62	88.00	6.20	20.00	9.09	0.25	0.06	76.0	15.2
RINSC	122	18 FLAT PLATES		7.62	100.33	5.50	19.90	18.79	0.28	0.26	6.0	1.2
SAPHIR (SWITZERLAND)	39	MTR PLATE TYPE	8.05	7.62	87.50	4.00	19.84	11.58	0.41	0.21	48.3	11.3
SEOUL 1&2 (SO. KOREA)	48	MULTI-PIN CLUSTER				4.30	20.00	8.68	0.25	0.09	62.0	12.4
SEOUL 1&2 (SO. KOREA)	120	MULTI-PIN CLUSTER				6.60	20.00	7.83	0.44	0.15	66.0	13.2
TR-2 (TURKEY)	9	MTR-C2 TYPE	7.62	7.62	88.00	5.60	20.00	9.09	0.31	0.12	60.0	12.0
TR-2 (TURKEY)	32	MTR-S TYPE	7.62	7.62	88.00	6.60	20.00	9.09	0.42	0.17	60.0	12.0
UMRR	28	24 CURVED PLATES	7.57	8.74	87.00	6.20	19.75	19.83	0.22	0.19	16.7	17.0
UNIV OF FLORIDA	25	11 FLAT PLATES	5.44	7.23	65.09	5.50	19.72	19.72	0.22	0.22	0.0	0.0
UNIV OF MASS-LOWELL	41	18 FLAT PLATES	7.62	7.62	101.60	8.86		19.73		0.26	0.0	0.0
UNIV OF MICHIGAN	330	18 CURVED PLATES	7.47	8.26	87.38	5.49	19.81	19.81	0.21	0.21	0.0	0.0
UNIV OF VIRGINIA	33	22 FLAT PLATES	8.26	7.61	93.88	5.80	19.75	19.03	0.28	0.23	14.6	11.4
TOTALS	6976					42149.33				1098.40		

A.4 Description of Al-SNF Materials

There are three basic fuel types that have been fabricated for research and test reactors. Originally fuel was made from cast aluminum-uranium alloys; later it was made using powder metallurgy techniques. These fuels include UAl_x, U_3O_8 and U_3Si_2 powders that are mixed with aluminum powder and hot/cold rolled to produce flat plates. Fuel elements are irradiated from 30 to 60% burnup.

A.4.1 U Oxide/Failed Clad & Al

A photomicrograph of the U_3O_8 fuel at high burnup is shown in Figure A.7. During irradiation, centerline temperatures up to about 200°C can occur which enhances diffusion of uranium and aluminum in the fuel meat. These effects produce chemical reactions between various fuel particles and matrix materials. For example, in U_3O_8 fuels a UAl_x type phase is formed. This phase can be seen in Figure A.7.

The kinetics of oxide dissolution in molten aluminum are slow because of the stability of the oxide phase, but the diffusion reactions in oxide fuels during irradiation will enhance the solubility of the fuel phase. During the MD treatment, the U_3O_8 fuel is expected to dissolve adequately when melted. For irradiated oxide fuels, the melting behavior is expected to be governed by the uranium-aluminum phase diagram shown in Figure A.8 of this report.

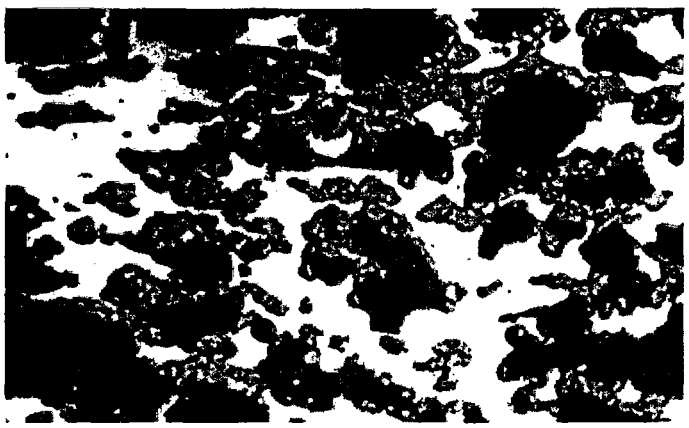


Figure A.7 High Burnup U_3O_8-Al Fuel Irradiated in Research and Test Reactors

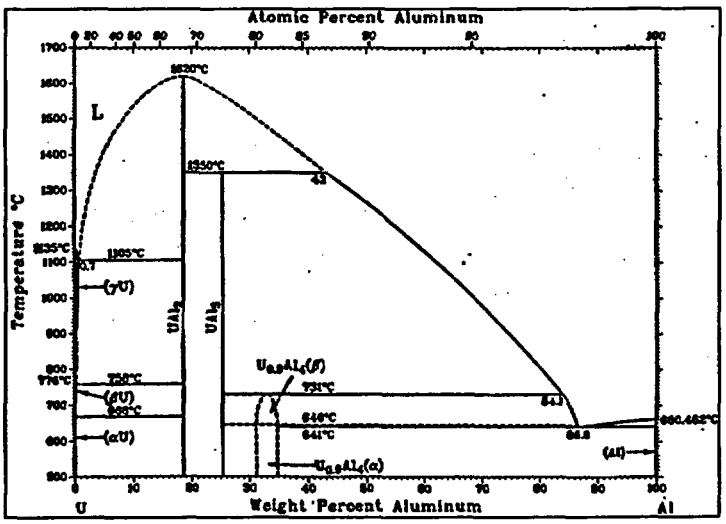


Figure A.8 Binary Phase Diagram of the Uranium-Aluminum System

A.4.2 UAl_x/Al

A photomicrograph of the UAl_x fuel at high burnup is shown in Figure A.9. During irradiation, centerline temperatures up to about 200°C can occur which enhances diffusion of uranium and aluminum in the fuel meat. For aluminide fuels, no reaction occurs at the particle-matrix boundary because of the thermodynamic stability of the aluminide phase present in the fuel as shown in Figure A.9.

During the MD treatment, the UAl_x fuel is expected to dissolve adequately when melted. For aluminide fuels, the melting behavior is expected to be governed by the uranium-aluminum phase diagram shown in Figure A.8 of this report.

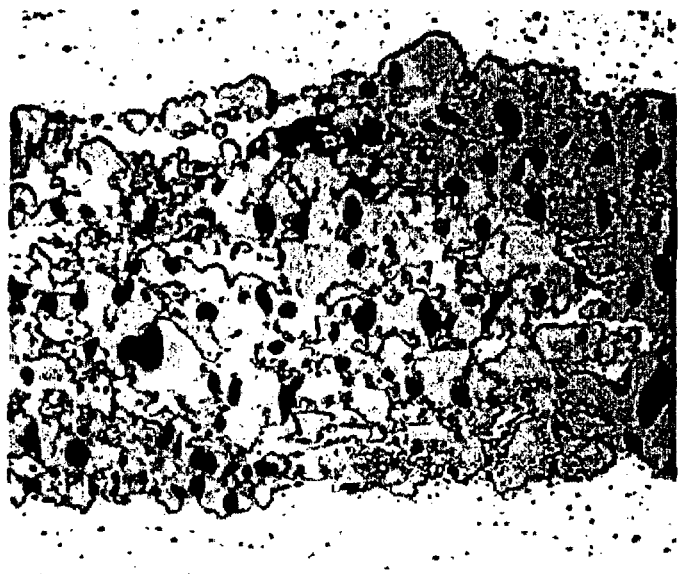


Figure A.9 High Burnup UAl_x-Al Fuel Irradiated in Research and Test Reactors

A.4.3 U-Si

A photomicrograph of the U_3Si_2 fuel at high burnup is shown in Figure A.10. During irradiation, centerline temperatures up to about 200°C can occur which enhances diffusion of uranium and aluminum in the fuel meat. These effects produce chemical reactions between various fuel particles and matrix materials. In silicide fuels aluminum-silicon phases form at the aluminum - U_3Si_2 particle interface. These phases can be seen in Figure A.10.

During the MD treatment, the U_3Si_2 fuel is expected to dissolve adequately when melted. For silicide fuels, the uranium-aluminum-silicon ternary phase diagram is necessary to predict process conditions. The ternary diagram at 950°C was constructed from binary phase diagrams and is shown in Figure A.11

Figure A.10 High Burnup U_3Si_2-Al Fuel Irradiated in Research and Test Reactors

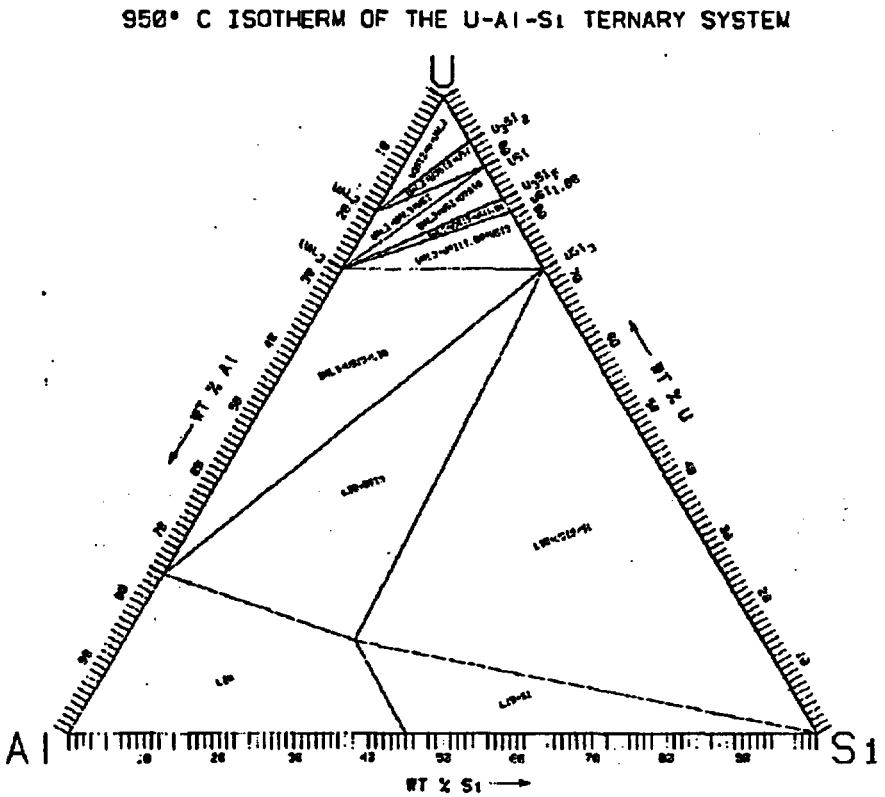


Figure A.11 Ternary Isothermal Section from the U-Al-Si System at 950°C

At 950°C the aluminum-uranium-silicon system has a relatively large liquidus region near the aluminum-rich end of the phase diagram. Calculations, based on aluminide fuels, for the silicide loading indicate that the melt composition will be in the range of about 1 to10 wt% uranium, 0.1-0.8 wt% silicon, and 98.9-89.2 wt% aluminum for various MTR assemblies. According to these calculations, the alloy is expected to melt between 660 and 960°C. Melting of silicide fuels can be accomplished using the same melt-dilute process; however, dilution may not be a concern for these low enriched elements. Melting and casting would, however, consolidate the fuel and produce a waste form consistent with other fuel types.

A.4.4 Sterling Forest Oxide-Type Material

This material consists of small particles of uranium-oxide fuel meat that has characteristics similar to that of Sterling Forest spent nuclear fuel.

A.4.5 Physical Condition of Fuel Assemblies

The SNF is typically stored under water where corrosion may be severe unless strict control of the water purity is maintained [2]. Several dry storage facilities are in use at foreign reactor sites. Prolonged underwater storage is not desirable because of the cost of operating and maintaining a properly controlled water system and the limited space and handling capabilities available at most reactor facilities.

An evaluation of the physical condition of a large portion of the fuel assemblies in wet and dry storage at foreign and domestic research reactor sites was recently performed [3]. Only minor corrosion and mechanical damage indicating cladding penetration was observed in approximately 7% of the 1700 fuel assemblies examined.

New criteria for acceptance of Al-SNF for SRS basin storage without special canning are provided in Reference 4. These criteria should be used for pre-treatment or pre-drying basin storage for the TSF if using the melt-dilute or direct disposal technologies, respectively.

A.4.6 Constituents of DOE SNF

In order to ensure the proper operation of the melt-dilute process and offgas system it is fundamentally necessary to compile a complete list of all the possible chemical species available for reaction. These various chemical species are present as radionuclides from the fission process, cladding alloy additions, fuel meat alloy additions/impurities, and also from any melt-dilute process additions such as the depleted uranium used for dilution of the ^{235}U content. Table A.6 contains a listing of the chemical species expected for the various cladding materials that will be processed in the melt-dilute process. Table A.7 lists the composition of aluminum powder used in UAl_x-fuel manufacturing. A chemical analysis of the depleted uranium used for dilution is provided in Table A.8.

Table A.6 Compositions of Typical MTR Cladding Alloys Manufactured by CERCA, NUKEM, and B&W

Elements	AG1	AG2	AG3	AlFeNi	AlMg1	AlMg2	6061
Mg	1.1-1.4	1.8-2.3	2.5-3.0	0.8-1.2	0.7-1.1	1.7-2.4	0.8-1.2
B	-	0.001	-	0.001	0.001	0.001	0.001
Cd	-	0.001	-	0.001	0.001	0.001	0.001
Cu	-	0.008	-	0.008	0.008	0.008	0.2-0.40
Fe	-	0.2-0.4	-	0.8-1.2	0.45	0.40	0.70
Si	-	0.3	-	0.30	0.30	0.30	0.40-0.80
Cr	-	0.3	-	0.2-0.5	0.10	0.30	0.04-0.35
Mn	-	0.7	-	0.2-0.6	0.15	0.30	0.15
Li	-	0.001	-	0.001	0.001	0.001	0.008
Zn	-	-	-	0.06-0.14	0.05	0.03	0.25
Ti	-	-	-	0.02-0.08	-	0.10	0.25
Al	-	Bal.	-	Bal.	Bal.	Bal.	Bal.
Others	-	0.03	-	0.03.	0.15	0.15	0.15

Table A.7 Composition of Al Powder Used for UAl_x Fuel Manufacturing

Element	Wt%
Al	99.00 min
Si+Fe	1.0 max
Cu	0.05-0.20
Mn	0.05 max
Zn	0.05 max
Other (each)	0.05 max
Others (total)	0.15 max

Table A.8 Chemical Analysis of the Depleted Uranium for Dilution

Element	Concentration, ppm
C	400
N_2	50
H_2	1
O_2	10
Al	< 6
B	< 0.15
Cd	< 20
Cr	10
Cu	3
Fe	125-225
Mg	< 4
Mn	5
Mo	< 6
Si	125-225
Sn	5
Zr	< 5

Table A.9 provides a bounding assessment of the fission product, actinide, and light element inventories expected in a single DOE SNF assembly. The table provides data for a low enriched (20%) and a highly enriched (93%) MTR assembly. The burnup was conservatively assumed to be 250 MWd/assembly. The final column of data (MAX) provides the maximum value of the other six columns. It is expected that the fission product inventory of any individual assembly processed by the MD technology option would be bounded by the high enrichment data provided in Table A.9, while the actinide inventory will be significantly bound by the low enrichment data. Actinide inventory data may be excessively conservative due to the assumed burnup, and consideration is ongoing.

Table A.9 Bounding Estimates of Mass of Fission Products, Actinides, and Light Elements (grams)

Radionuclide/ Element	Low Enrichment			High Enrichment			MAX
	initial	5.0 yr	10.0 yr	initial	5.0 yr	10.0 yr	
Ac225	1.28E-16	3.13E-16	1.21E-15	9.00E-18	5.99E-16	1.34E-15	1.34E-15
Ac227	1.13E-14	1.15E-13	2.96E-13	1.48E-13	1.71E-10	6.33E-10	6.33E-10
Actinium	**1.14E-14**	**1.15E-13**	**2.97E-13**	**1.48E-13**	**1.71E-10**	**6.33E-10**	**6.33E-10**
Al27	1.03E+03	1.03E+03	1.03E+03	1.03E+03	1.03E+03	1.03E+03	1.03E+03
Aluminum	**1.03E+03**	**1.03E+03**	**1.03E+03**	**1.03E+03**	**1.03E+03**	**1.03E+03**	**1.03E+03**
Am241	6.84E-04	4.79E-01	8.51E-01	9.99E-05	1.48E-02	2.61E-02	8.51E-01
Am242	1.03E-04	1.18E-10	1.15E-10	1.18E-06	1.44E-11	1.40E-11	1.03E-04
Am242m	9.34E-06	9.11E-06	8.89E-06	1.14E-06	1.12E-06	1.09E-06	9.34E-06
Am243	1.61E+00	1.70E+00	1.70E+00	1.83E-04	1.89E-04	1.88E-04	1.70E+00
Am245	1.41E-16	2.71E-18	5.20E-20	0.00E+00	0.00E+00	0.00E+00	1.41E-16
Americium	**1.61E+00**	**2.18E+00**	**2.55E+00**	**2.85E-04**	**1.49E-02**	**2.63E-02**	**2.55E+00**
Sb121	2.40E-02	2.53E-02	2.53E-02	1.71E-02	1.78E-02	1.78E-02	2.53E-02
Sb123	2.71E-02	2.95E-02	2.95E-02	2.00E-02	2.23E-02	2.23E-02	2.95E-02
Sb124	1.06E-03	7.79E-13	5.73E-22	7.18E-05	5.28E-14	3.89E-23	1.06E-03
Sb125	5.79E-02	1.80E-02	5.08E-03	3.78E-02	1.16E-02	3.26E-03	5.79E-02
Sb126	5.20E-04	5.64E-09	5.64E-09	6.54E-05	3.00E-09	3.00E-09	5.20E-04
Sb126m	1.19E-06	4.29E-11	4.29E-11	9.89E-08	2.28E-11	2.28E-11	1.19E-06
Antimony	**1.11E-01**	**7.28E-02**	**5.98E-02**	**7.50E-02**	**5.16E-02**	**4.33E-02**	**1.11E-01**
As75	8.49E-04	8.50E-04	8.50E-04	1.00E-03	1.00E-03	1.00E-03	1.00E-03
Arsenic	**8.49E-04**	**8.50E-04**	**8.50E-04**	**1.00E-03**	**1.00E-03**	**1.00E-03**	**1.00E-03**
Ba132	8.66E-07	1.58E-06	1.58E-06	1.76E-07	2.73E-07	2.73E-07	1.58E-06
Ba133	6.00E-10	4.31E-10	3.10E-10	8.06E-11	5.80E-11	4.18E-11	6.00E-10
Ba134	1.50E-02	1.08E+00	1.28E+00	3.16E-03	2.01E-01	2.38E-01	1.28E+00
Ba135	1.63E-04	1.70E-04	1.73E-04	3.20E-06	4.19E-06	5.14E-06	1.73E-04
Ba136	2.70E-02	8.53E-02	8.53E-02	7.65E-03	1.28E-02	1.28E-02	8.53E-02
Ba136m	1.80E-09	0.00E+00	0.00E+00	1.56E-10	0.00E+00	0.00E+00	1.80E-09
Ba137	1.80E-02	1.08E+00	2.01E+00	1.70E-02	1.08E+00	2.04E+00	2.04E+00
Ba137m	1.70E-06	1.31E-06	1.17E-06	1.55E-06	1.33E-06	1.19E-06	1.70E-06
Ba138	9.91E+00	9.93E+00	9.93E+00	1.05E+01	1.05E+01	1.05E+01	1.05E+01
Ba140	3.11E+00	0.00E+00	0.00E+00	3.43E+00	0.00E+00	0.00E+00	3.43E+00
Barium	**1.31E+01**	**1.22E+01**	**1.33E+01**	**1.40E+01**	**1.18E+01**	**1.28E+01**	**1.40E+01**
Bk249	3.51E-08	7.14E-10	1.36E-11	9.15E-18	1.79E-19	3.41E-21	3.51E-08
Bk250	3.96E-09	2.71E-17	2.94E-19	0.00E+00	0.00E+00	0.00E+00	3.96E-09

Radionuclide/ Element	Low Enrichment			High Enrichment			MAX
	initial	5.0 yr	10.0 yr	initial	5.0 yr	10.0 yr	
Bk251	4.16E-11	2.69E-30	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.16E-11
Berkelium	**3.91E-08**	**7.14E-10**	**1.36E-11**	**9.15E-18**	**1.79E-19**	**3.41E-21**	**3.91E-08**
Be9	1.49E-07	1.49E-07	1.49E-07	1.54E-07	1.54E-07	1.54E-07	1.54E-07
Be10	9.90E-07	9.90E-07	9.90E-07	1.02E-06	1.02E-06	1.02E-06	1.02E-06
Beryllium	**1.14E-06**	**1.14E-06**	**1.14E-06**	**1.18E-06**	**1.18E-06**	**1.18E-06**	**1.18E-06**
Bi209	4.80E-17	1.31E-14	9.64E-14	5.06E-18	3.35E-14	1.45E-13	1.45E-13
Bi210	2.94E-20	8.23E-19	8.18E-18	0.00E+00	0.00E+00	0.00E+00	8.18E-18
Bi211	0.00E+00	0.00E+00	0.00E+00	5.55E-21	2.98E-17	1.10E-16	1.10E-16
Bi212	1.21E-16	2.05E-12	3.63E-12	2.68E-16	1.74E-13	2.94E-13	3.63E-12
Bi213	0.00E+00	0.00E+00	0.00E+00	2.70E-20	1.80E-18	4.03E-18	4.03E-18
Bismuth	**1.69E-16**	**2.06E-12**	**3.72E-12**	**2.73E-16**	**2.07E-13**	**4.39E-13**	**3.72E-12**
Br79	1.06E-08	3.85E-07	7.59E-07	9.64E-09	4.28E-07	8.46E-07	8.46E-07
Br81	1.53E-01	1.53E-01	1.53E-01	1.76E-01	1.76E-01	1.76E-01	1.76E-01
Bromine	**1.53E-01**	**1.53E-01**	**1.53E-01**	**1.76E-01**	**1.76E-01**	**1.76E-01**	**1.76E-01**
Cd108	1.24E-06	1.24E-06	1.24E-06	2.29E-08	2.29E-08	2.29E-08	1.24E-06
Cd109	1.07E-09	6.95E-11	4.50E-12	7.20E-12	4.68E-13	3.03E-14	1.07E-09
Cd110	4.31E-01	4.49E-01	4.49E-01	5.11E-03	5.30E-03	5.31E-03	4.49E-01
Cd111	9.55E-02	1.61E-01	1.61E-01	2.04E-02	2.59E-02	2.59E-02	1.61E-01
Cd112	8.44E-02	8.94E-02	8.94E-02	2.03E-02	2.09E-02	2.09E-02	8.94E-02
Cd113	7.49E-05	6.74E-04	6.74E-04	6.71E-04	7.95E-04	7.95E-04	7.95E-04
Cd113m	1.05E-03	8.29E-04	6.48E-04	3.49E-04	2.74E-04	2.15E-04	1.05E-03
Cd114	8.33E-02	8.33E-02	8.33E-02	3.85E-02	3.85E-02	3.85E-02	8.33E-02
Cd115m	6.75E-04	3.18E-16	1.49E-28	4.16E-04	1.95E-16	9.15E-29	6.75E-04
Cd116	3.19E-02	3.19E-02	3.19E-02	2.24E-02	2.24E-02	2.24E-02	3.19E-02
Cadmium	**7.28E-01**	**8.16E-01**	**8.16E-01**	**1.08E-01**	**1.14E-01**	**1.14E-01**	**8.16E-01**
Cf249	6.55E-11	3.64E-08	3.68E-08	6.64E-20	9.13E-18	9.21E-18	3.68E-08
Cf250	1.76E-08	1.66E-08	1.28E-08	5.41E-19	4.40E-19	3.38E-19	1.76E-08
Cf251	6.00E-09	6.01E-09	5.99E-09	0.00E+00	0.00E+00	0.00E+00	6.01E-09
Cf252	1.68E-08	4.53E-09	1.22E-09	0.00E+00	0.00E+00	0.00E+00	1.68E-08
Cf253	2.91E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.91E-10
Cf254	3.40E-12	2.78E-21	2.28E-30	0.00E+00	0.00E+00	0.00E+00	3.40E-12
Californium	**4.07E-08**	**6.35E-08**	**5.67E-08**	**6.08E-19**	**9.57E-18**	**9.55E-18**	**6.35E-08**
Ca46	4.08E-19	4.08E-19	4.08E-19	0.00E+00	0.00E+00	0.00E+00	4.08E-19
Calcium	**4.08E-19**	**4.08E-19**	**4.08E-19**	**0.00E+00**	**0.00E+00**	**0.00E+00**	**4.08E-19**
C14	2.00E-07	2.00E-07	2.00E-07	2.08E-07	2.08E-07	2.06E-07	2.08E-07
Carbon	**2.00E-07**	**2.00E-07**	**2.00E-07**	**2.08E-07**	**2.08E-07**	**2.06E-07**	**2.08E-07**
Ce139	3.38E-07	3.41E-11	3.45E-15	9.19E-08	9.30E-12	9.40E-16	3.38E-07
Ce140	6.44E+00	1.01E+01	1.01E+01	6.20E+00	1.01E+01	1.01E+01	1.01E+01
Ce141	4.43E+00	5.40E-17	6.54E-34	5.64E+00	6.88E-17	8.33E-34	5.64E+00
Ce142	9.86E+00	9.88E+00	9.88E+00	9.53E+00	9.55E+00	9.55E+00	9.88E+00
Ce144	7.40E+00	8.70E-02	1.02E-03	8.44E+00	9.93E-02	1.17E-03	8.44E+00
Cerium	**2.81E+01**	**2.00E+01**	**1.99E+01**	**2.98E+01**	**1.97E+01**	**1.96E+01**	**2.98E+01**
Cs133	4.83E+00	5.91E+00	5.91E+00	8.24E+00	9.90E+00	9.90E+00	9.90E+00
Cs134	1.30E+00	2.43E-01	4.51E-02	2.43E-01	4.53E-02	8.43E-03	1.30E+00
Cs135	1.10E+00	1.17E+00	1.17E+00	5.50E-01	6.28E-01	6.28E-01	1.17E+00
Cs136	5.83E-02	0.00E+00	0.00E+00	5.13E-03	0.00E+00	0.00E+00	5.83E-02
Cs137	9.69E+00	8.63E+00	7.69E+00	9.78E+00	8.71E+00	7.76E+00	9.78E+00
Cesium	**1.70E+01**	**1.60E+01**	**1.48E+01**	**1.88E+01**	**1.93E+01**	**1.83E+01**	**1.93E+01**
Cr52	6.19E-06	6.20E-06	6.20E-06	1.25E-06	1.25E-06	1.25E-06	6.20E-06

Radionuclide/ Element	Low Enrichment			High Enrichment			MAX
	initial	5.0 yr	10.0 yr	initial	5.0 yr	10.0 yr	
Cr53	3.30E-07	3.30E-07	3.30E-07	4.90E-09	4.90E-09	4.90E-09	3.30E-07
Cr54	8.14E-07	1.35E-05	1.38E-05	2.11E-07	2.81E-06	2.86E-06	1.38E-05
Chromium	**7.33E-06**	**2.00E-05**	**2.03E-05**	**1.46E-06**	**4.06E-06**	**4.11E-06**	**2.03E-05**
Co58	9.29E-16	1.66E-23	2.94E-31	0.00E+00	0.00E+00	0.00E+00	9.29E-16
Co59	1.44E-08	2.65E-07	2.65E-07	3.05E-13	2.48E-12	2.48E-12	2.65E-07
Co60	1.13E-09	5.88E-10	3.04E-10	1.64E-15	8.50E-16	4.40E-16	1.13E-09
Cobalt	**1.55E-08**	**2.66E-07**	**2.65E-07**	**3.07E-13**	**2.48E-12**	**2.48E-12**	**2.66E-07**
Cu63	1.13E-20	9.16E-18	1.80E-17	0.00E+00	0.00E+00	0.00E+00	1.80E-17
Copper	**1.13E-20**	**9.16E-18**	**1.80E-17**	**0.00E+00**	**0.00E+00**	**0.00E+00**	**1.80E-17**
Cm241	1.30E-09	2.25E-26	0.00E+00	4.90E-13	8.48E-30	0.00E+00	1.30E-09
Cm242	1.65E-03	7.59E-07	2.35E-08	9.25E-06	7.24E-09	2.84E-09	1.65E-03
Cm243	4.98E-05	4.41E-05	3.90E-05	4.88E-08	4.31E-08	3.83E-08	4.98E-05
Cm244	5.83E-01	5.33E-01	4.40E-01	8.66E-06	7.75E-06	6.40E-06	5.83E-01
Cm245	1.24E-02	1.24E-02	1.24E-02	9.44E-08	9.43E-08	9.43E-08	1.24E-02
Cm246	8.80E-03	8.79E-03	8.79E-03	1.98E-09	1.98E-09	1.98E-09	8.80E-03
Cm247	5.53E-05	5.53E-05	5.53E-05	1.63E-12	1.63E-12	1.63E-12	5.53E-05
Cm248	6.30E-06	6.31E-06	6.33E-06	9.60E-15	9.60E-15	9.60E-15	6.33E-06
Cm250	6.43E-12	6.44E-12	6.43E-12	0.00E+00	0.00E+00	0.00E+00	6.44E-12
Curium	**6.05E-01**	**5.54E-01**	**4.61E-01**	**1.81E-05**	**7.90E-06**	**6.54E-06**	**6.05E-01**
Dy160	6.75E-05	1.31E-03	1.31E-03	3.44E-06	2.59E-05	2.59E-05	1.31E-03
Dy161	7.63E-04	3.68E-03	3.68E-03	1.21E-04	1.59E-04	1.59E-04	3.68E-03
Dy162	1.49E-03	1.49E-03	1.49E-03	5.73E-05	5.73E-05	5.73E-05	1.49E-03
Dy163	1.31E-03	1.31E-03	1.31E-03	2.13E-05	2.13E-05	2.13E-05	1.31E-03
Dy164	9.88E-05	9.89E-05	9.89E-05	4.25E-06	4.25E-06	4.25E-06	9.89E-05
Dysprosium	**3.73E-03**	**7.89E-03**	**7.89E-03**	**2.07E-04**	**2.67E-04**	**2.67E-04**	**7.89E-03**
Es253	1.56E-11	3.14E-36	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.56E-11
Es254	5.60E-12	5.68E-14	5.75E-16	0.00E+00	0.00E+00	0.00E+00	5.60E-12
Es255	3.21E-13	2.69E-27	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.21E-13
Einsteinium	**2.15E-11**	**5.68E-14**	**5.75E-16**	**0.00E+00**	**0.00E+00**	**0.00E+00**	**2.15E-11**
Er166	2.34E-04	3.94E-04	3.94E-04	3.98E-07	4.61E-07	4.61E-07	3.94E-04
Er167	6.89E-06	6.89E-06	6.89E-06	2.95E-09	2.95E-09	2.95E-09	6.89E-06
Er168	2.14E-05	2.14E-05	2.14E-05	8.06E-10	8.06E-10	8.06E-10	2.14E-05
Er170	1.96E-08	1.96E-08	1.96E-08	5.49E-10	5.49E-10	5.49E-10	1.96E-08
Erbium	**2.62E-04**	**4.22E-04**	**4.22E-04**	**4.02E-07**	**4.66E-07**	**4.66E-07**	**4.22E-04**
Eu149	1.08E-12	1.33E-18	1.64E-24	4.26E-14	5.26E-20	6.50E-26	1.08E-12
Eu150	8.54E-13	7.75E-13	7.04E-13	4.56E-11	4.14E-11	3.76E-11	4.56E-11
Eu151	1.22E-07	1.81E-03	3.56E-03	3.33E-05	5.85E-03	1.14E-02	1.14E-02
Eu152	1.30E-06	1.00E-06	7.71E-07	4.41E-05	3.41E-05	2.63E-05	4.41E-05
Eu153	5.61E-01	8.91E-01	8.91E-01	4.03E-01	4.44E-01	4.44E-01	8.91E-01
Eu154	1.04E-01	6.91E-02	4.63E-02	4.04E-02	2.70E-02	1.80E-02	1.04E-01
Eu155	3.61E-02	1.74E-02	8.26E-03	1.48E-02	7.05E-03	3.36E-03	3.61E-02
Eu156	3.85E-01	2.41E-37	0.00E+00	3.64E-02	0.00E+00	0.00E+00	3.85E-01
Europium	**1.09E+00**	**9.80E-01**	**9.49E-01**	**4.94E-01**	**4.84E-01**	**4.77E-01**	**1.09E+00**
F19	2.33E-17	2.33E-17	2.33E-17	0.00E+00	0.00E+00	0.00E+00	2.33E-17
Fluorine	**2.33E-17**	**2.33E-17**	**2.33E-17**	**0.00E+00**	**0.00E+00**	**0.00E+00**	**2.33E-17**
Fr223	0.00E+00	0.00E+00	0.00E+00	3.81E-21	4.41E-18	1.64E-17	1.64E-17
Francium	**0.00E+00**	**0.00E+00**	**0.00E+00**	**3.81E-21**	**4.41E-18**	**1.64E-17**	**1.64E-17**
Gd152	4.05E-06	4.45E-06	4.51E-06	2.23E-05	2.55E-05	2.78E-05	2.78E-05
Gd153	5.83E-06	3.09E-08	1.64E-10	7.28E-07	3.85E-09	2.04E-11	5.83E-06

Radionuclide/ Element	Low Enrichment			High Enrichment			MAX
	initial	5.0 yr	10.0 yr	initial	5.0 yr	10.0 yr	
Gd154	2.78E-04	3.46E-02	5.76E-02	1.34E-04	1.35E-02	2.25E-02	5.76E-02
Gd155	1.15E-06	1.90E-02	2.81E-02	1.24E-05	7.75E-03	1.14E-02	2.81E-02
Gd156	2.54E-01	6.41E-01	6.41E-01	4.30E-02	7.96E-02	7.96E-02	6.41E-01
Gd157	1.28E-03	8.76E-02	8.76E-02	2.83E-04	7.06E-04	7.06E-04	8.76E-02
Gd158	1.26E+00	1.26E+00	1.26E+00	2.21E-02	2.21E-02	2.21E-02	1.26E+00
Gd160	6.78E-03	6.78E-03	6.78E-03	6.10E-04	6.10E-04	6.10E-04	6.78E-03
Gadolinium	**1.52E+00**	**2.05E+00**	**2.08E+00**	**6.62E-02**	**1.24E-01**	**1.37E-01**	**2.08E+00**
Ga69	8.26E-10	8.83E-10	8.83E-10	5.79E-10	5.86E-10	5.86E-10	8.83E-10
Ga71	2.36E-07	2.40E-07	2.40E-07	1.96E-07	1.96E-07	1.96E-07	2.40E-07
Gallium	**2.37E-07**	**2.41E-07**	**2.41E-07**	**1.97E-07**	**1.97E-07**	**1.97E-07**	**2.41E-07**
Ge70	2.31E-11	2.33E-11	2.33E-11	9.46E-12	9.46E-12	9.46E-12	2.33E-11
Ge72	2.35E-05	2.45E-05	2.45E-05	2.23E-05	2.38E-05	2.38E-05	2.45E-05
Ge73	6.90E-05	6.98E-05	6.98E-05	9.69E-05	9.75E-05	9.75E-05	9.75E-05
Ge74	7.74E-05	7.74E-05	7.74E-05	8.65E-05	8.66E-05	8.66E-05	8.66E-05
Ge76	2.79E-03	2.79E-03	2.79E-03	3.34E-03	3.34E-03	3.34E-03	3.34E-03
Germanium	**2.96E-03**	**2.96E-03**	**2.96E-03**	**3.54E-03**	**3.55E-03**	**3.55E-03**	**3.55E-03**
He3	9.33E-12	3.13E-08	5.48E-08	4.81E-11	5.85E-09	1.02E-08	5.48E-08
He4	1.07E-03	3.34E-03	5.29E-03	2.00E-04	2.40E-04	2.80E-04	5.29E-03
Helium	**1.07E-03**	**3.34E-03**	**5.29E-03**	**2.00E-04**	**2.40E-04**	**2.80E-04**	**5.29E-03**
Ho165	5.40E-04	5.54E-04	5.54E-04	5.28E-06	5.30E-06	5.30E-06	5.54E-04
Ho166m	2.36E-06	2.36E-06	2.35E-06	4.73E-09	4.71E-09	4.70E-09	2.36E-06
Holmium	**5.42E-04**	**5.56E-04**	**5.56E-04**	**5.28E-06**	**5.30E-06**	**5.30E-06**	**5.56E-04**
H1	1.61E-03	1.61E-03	1.61E-03	3.08E-04	3.08E-04	3.08E-04	1.61E-03
H2	1.48E-05	1.48E-05	1.48E-05	2.25E-06	2.25E-06	2.25E-06	1.48E-05
H3	3.90E-04	2.95E-04	2.23E-04	3.68E-04	2.78E-04	2.10E-04	3.90E-04
Hydrogen	**2.02E-03**	**1.92E-03**	**1.85E-03**	**6.77E-04**	**5.87E-04**	**5.20E-04**	**2.02E-03**
In113	1.85E-06	2.33E-04	4.13E-04	1.16E-06	7.75E-05	1.38E-04	4.13E-04
In114	6.70E-12	1.26E-23	9.91E-35	1.86E-13	4.79E-25	0.00E+00	6.70E-12
In114m	9.95E-08	7.84E-19	6.16E-30	3.79E-09	2.98E-20	2.34E-31	9.95E-08
In115	5.63E-03	9.09E-03	9.09E-03	1.15E-02	1.29E-02	1.29E-02	1.29E-02
In115m	2.26E-04	1.46E-22	0.00E+00	7.39E-05	9.04E-23	0.00E+00	2.26E-04
Indium	**5.85E-03**	**9.32E-03**	**9.50E-03**	**1.16E-02**	**1.30E-02**	**1.30E-02**	**1.30E-02**
I127	1.96E-01	2.90E-01	2.90E-01	1.39E-01	1.85E-01	1.85E-01	2.90E-01
I129	9.23E-01	1.09E+00	1.09E+00	9.61E-01	1.09E+00	1.09E+00	1.09E+00
I131	1.09E+00	0.00E+00	0.00E+00	9.73E-01	0.00E+00	0.00E+00	1.09E+00
Iodine	**2.21E+00**	**1.38E+00**	**1.38E+00**	**2.07E+00**	**1.28E+00**	**1.28E+00**	**2.21E+00**
Fe56	7.03E-01	7.08E-01	7.08E-01	4.70E-02	4.73E-02	4.73E-02	7.08E-01
Fe57	9.88E-03	9.88E-03	9.88E-03	3.36E-05	3.36E-05	3.36E-05	9.88E-03
Fe58	8.65E-05	8.65E-05	8.65E-05	1.51E-08	1.51E-08	1.51E-08	8.65E-05
Fe59	2.51E-07	1.11E-19	4.90E-32	2.18E-12	9.61E-25	4.25E-37	2.51E-07
Iron	**7.12E-01**	**7.17E-01**	**7.17E-01**	**4.70E-02**	**4.73E-02**	**4.73E-02**	**7.17E-01**
Kr80	1.14E-07	1.17E-07	1.17E-07	6.70E-08	6.74E-08	6.74E-08	1.17E-07
Kr81	2.41E-08	2.41E-08	2.41E-08	3.85E-09	3.85E-09	3.85E-09	2.41E-08
Kr82	4.03E-03	4.70E-03	4.70E-03	3.59E-04	3.90E-04	3.90E-04	4.70E-03
Kr83	8.68E-02	8.80E-02	8.80E-02	4.73E-01	4.75E-01	4.75E-01	4.75E-01
Kr84	1.16E+00	1.16E+00	1.16E+00	1.05E+00	1.05E+00	1.05E+00	1.16E+00
Kr85	2.01E-01	1.46E-01	1.06E-01	2.55E-01	1.85E-01	1.34E-01	2.55E-01
Kr86	1.51E+00	1.51E+00	1.51E+00	1.93E+00	1.93E+00	1.93E+00	1.93E+00
Krypton	**2.97E+00**	**2.91E+00**	**2.87E+00**	**3.71E+00**	**3.64E+00**	**3.59E+00**	**3.71E+00**

Radionuclide/ Element	Low Enrichment			High Enrichment			MAX
	Initial	5.0 yr	10.0 yr	Initial	5.0 yr	10.0 yr	
La138	5.96E-05	5.96E-05	5.96E-05	5.48E-05	5.48E-05	5.48E-05	5.96E-05
La139	8.79E+00	8.80E+00	8.80E+00	1.00E+01	1.00E+01	1.00E+01	1.00E+01
La140	5.20E-01	0.00E+00	0.00E+00	4.51E-01	0.00E+00	0.00E+00	5.20E-01
Lanthanum	**9.31E+00**	**8.80E+00**	**8.80E+00**	**1.05E+01**	**1.00E+01**	**1.00E+01**	**1.05E+01**
Pb206	6.60E-22	8.13E-17	8.58E-16	3.50E-22	7.35E-17	1.34E-15	1.34E-15
Pb207	4.35E-18	7.96E-15	3.66E-14	2.99E-17	8.39E-12	6.29E-11	6.29E-11
Pb208	2.54E-14	2.59E-08	1.13E-07	4.78E-14	2.33E-09	9.50E-09	1.13E-07
Pb209	1.60E-18	3.94E-18	1.53E-17	1.13E-19	7.54E-18	1.69E-17	1.69E-17
Pb210	9.94E-17	1.34E-15	1.33E-14	2.26E-17	2.48E-15	1.91E-14	1.91E-14
Pb211	0.00E+00	0.00E+00	0.00E+00	9.35E-20	5.03E-16	1.85E-15	1.85E-15
Pb212	1.28E-15	2.16E-11	3.83E-11	2.81E-15	1.83E-12	3.10E-12	3.83E-11
Lead	**2.68E-14**	**2.59E-08**	**1.13E-07**	**5.06E-14**	**2.34E-09**	**9.57E-09**	**1.13E-07**
Li6	1.20E-07	1.20E-07	1.20E-07	2.66E-06	2.66E-06	2.66E-06	2.66E-06
Li7	7.71E-08	7.71E-08	7.71E-08	7.96E-08	7.96E-08	7.96E-08	7.96E-08
Lithium	**1.97E-07**	**1.97E-07**	**1.97E-07**	**2.74E-06**	**2.74E-06**	**2.74E-06**	**2.74E-06**
Mg24	6.10E-03	6.39E-03	6.39E-03	1.17E-03	1.20E-03	1.20E-03	6.39E-03
Mg25	5.13E-06	5.13E-06	5.13E-06	2.40E-07	2.40E-07	2.40E-07	5.13E-06
Mg26	1.50E-04	1.50E-04	1.50E-04	2.79E-05	2.79E-05	2.79E-05	1.50E-04
Magnesium	**6.26E-03**	**6.54E-03**	**6.54E-03**	**1.20E-03**	**1.23E-03**	**1.23E-03**	**6.54E-03**
Mn54	1.30E-05	2.25E-07	3.89E-09	2.65E-06	4.59E-08	7.95E-10	1.30E-05
Mn55	4.46E+00	4.46E+00	4.46E+00	5.13E+00	5.13E+00	5.13E+00	5.13E+00
Manganese	**4.46E+00**	**4.46E+00**	**4.46E+00**	**5.13E+00**	**5.13E+00**	**5.13E+00**	**5.13E+00**
Mo95	3.69E-01	6.13E+00	6.13E+00	4.34E-01	7.01E+00	7.01E+00	7.01E+00
Mo96	6.39E-02	6.56E-02	6.56E-02	4.56E-03	4.65E-03	4.65E-03	6.56E-02
Mo97	5.74E+00	5.84E+00	5.84E+00	6.09E+00	6.23E+00	6.23E+00	6.23E+00
Mo98	6.29E+00	6.29E+00	6.29E+00	6.45E+00	6.45E+00	6.45E+00	6.45E+00
Mo100	7.00E+00	7.00E+00	7.00E+00	7.10E+00	7.10E+00	7.10E+00	7.10E+00
Molybdenum	**1.95E+01**	**2.53E+01**	**2.53E+01**	**2.01E+01**	**2.68E+01**	**2.68E+01**	**2.68E+01**
Nd142	2.26E-01	2.50E-01	2.50E-01	1.21E-02	1.29E-02	1.29E-02	2.50E-01
Nd143	7.59E-01	2.85E+00	2.85E+00	5.35E+00	9.08E+00	9.08E+00	9.08E+00
Nd144	6.59E+00	1.39E+01	1.40E+01	1.11E+00	9.44E+00	9.54E+00	1.40E+01
Nd145	3.94E+00	3.96E+00	3.96E+00	6.24E+00	6.29E+00	6.29E+00	6.29E+00
Nd146	6.69E+00	6.69E+00	6.69E+00	5.11E+00	5.13E+00	5.13E+00	6.69E+00
Nd147	6.99E-01	0.00E+00	0.00E+00	1.10E+00	0.00E+00	0.00E+00	1.10E+00
Nd148	3.46E+00	3.46E+00	3.46E+00	2.90E+00	2.90E+00	2.90E+00	3.46E+00
Nd150	1.26E+00	1.26E+00	1.26E+00	1.11E+00	1.11E+00	1.11E+00	1.26E+00
Neodymium	**2.36E+01**	**3.24E+01**	**3.25E+01**	**2.29E+01**	**3.39E+01**	**3.40E+01**	**3.40E+01**
Ne20	3.36E-16	3.36E-16	3.36E-16	2.45E-18	2.45E-18	2.45E-18	3.36E-16
Ne21	1.09E-08	1.09E-08	1.09E-08	4.11E-10	4.11E-10	4.11E-10	1.09E-08
Ne22	1.03E-10	1.03E-10	1.03E-10	9.53E-13	9.53E-13	9.53E-13	1.03E-10
Neon	**1.10E-08**	**1.10E-08**	**1.10E-08**	**4.12E-10**	**4.12E-10**	**4.12E-10**	**1.10E-08**
Np235	9.46E-08	3.88E-09	1.59E-10	4.93E-09	2.01E-10	8.26E-12	9.46E-08
Np236	3.30E-06	3.29E-06	3.29E-06	1.74E-07	1.74E-07	1.74E-07	3.30E-06
Np237	1.81E+00	4.98E+00	4.98E+00	8.33E-01	1.28E+00	1.28E+00	4.98E+00
Np238	2.04E-01	1.66E-12	1.61E-12	1.02E-02	2.03E-13	1.99E-13	2.04E-01
Np239	1.48E+01	1.46E-06	1.46E-06	1.84E-01	1.63E-10	1.61E-10	1.48E+01
Neptunium	**1.68E+01**	**4.98E+00**	**4.98E+00**	**1.03E+00**	**1.28E+00**	**1.28E+00**	**1.68E+01**
Ni60	2.70E-12	5.49E-10	8.31E-10	6.00E-18	7.96E-16	1.21E-15	8.31E-10
Ni61	4.04E-12	4.16E-12	4.16E-12	3.18E-19	3.23E-19	3.23E-19	4.16E-12

Radionuclide/ Element	Low Enrichment			High Enrichment			MAX
	initial	5.0 yr	10.0 yr	initial	5.0 yr	10.0 yr	
Ni62	1.46E-14	1.46E-14	1.46E-14	0.00E+00	0.00E+00	0.00E+00	1.46E-14
Ni63	2.69E-16	2.60E-16	2.51E-16	0.00E+00	0.00E+00	0.00E+00	2.69E-16
Ni64	7.33E-18	7.33E-18	7.33E-18	0.00E+00	0.00E+00	0.00E+00	7.33E-18
Nickel	**6.75E-12**	**5.53E-10**	**8.35E-10**	**6.32E-18**	**7.97E-16**	**1.21E-15**	**8.35E-10**
Nb91	6.21E-12	6.18E-12	6.15E-12	2.88E-12	2.85E-12	2.84E-12	6.21E-12
Nb92	7.70E-10	7.70E-10	7.70E-10	3.88E-10	3.88E-10	3.88E-10	7.70E-10
Nb93	6.34E-08	9.25E-07	3.24E-06	3.50E-08	1.07E-06	3.86E-06	3.86E-06
Nb93m	1.41E-07	7.61E-06	1.36E-05	1.45E-07	9.20E-06	1.65E-05	1.65E-05
Nb94	2.19E-06	2.19E-06	2.19E-06	1.79E-06	1.79E-06	1.79E-06	2.19E-06
Nb95	1.07E+00	1.45E-08	3.75E-17	1.17E+00	1.68E-08	4.33E-17	1.17E+00
Nb95m	2.79E-03	8.01E-12	2.06E-20	3.11E-03	9.25E-12	2.39E-20	3.11E-03
Niobium	**1.07E+00**	**1.07E-05**	**1.91E-05**	**1.17E+00**	**1.21E-05**	**2.22E-05**	**1.17E+00**
N15	4.03E-17	4.03E-17	4.03E-17	0.00E+00	0.00E+00	0.00E+00	4.03E-17
Nitrogen	**4.03E-17**	**4.03E-17**	**4.03E-17**	**0.00E+00**	**0.00E+00**	**0.00E+00**	**4.03E-17**
O18	4.75E-11	4.75E-11	4.75E-11	8.86E-14	8.86E-14	8.86E-14	4.75E-11
Oxygen	**4.75E-11**	**4.75E-11**	**4.75E-11**	**8.86E-14**	**8.86E-14**	**8.86E-14**	**4.75E-11**
Pd104	9.16E-01	9.16E-01	9.16E-01	9.76E-02	9.76E-02	9.76E-02	9.16E-01
Pd105	3.03E-01	3.46E-01	3.46E-01	7.76E-01	8.14E-01	8.14E-01	8.14E-01
Pd106	2.51E+00	4.40E+00	4.46E+00	3.79E-01	8.36E-01	8.51E-01	4.46E+00
Pd107	1.36E+00	1.36E+00	1.36E+00	1.78E-01	1.78E-01	1.78E-01	1.36E+00
Pd108	9.65E-01	9.65E-01	9.65E-01	8.75E-02	8.75E-02	8.75E-02	9.65E-01
Pd110	2.98E-01	2.98E-01	2.98E-01	3.95E-02	3.95E-02	3.95E-02	2.98E-01
Palladium	**6.36E+00**	**8.29E+00**	**8.35E+00**	**1.56E+00**	**2.05E+00**	**2.07E+00**	**8.35E+00**
P31	6.56E-10	6.79E-10	6.79E-10	5.16E-15	5.24E-15	5.24E-15	6.79E-10
P32	3.23E-13	5.91E-18	5.79E-18	0.00E+00	0.00E+00	0.00E+00	3.23E-13
Phosphorus	**6.57E-10**	**6.79E-10**	**6.79E-10**	**5.16E-15**	**5.24E-15**	**5.24E-15**	**6.79E-10**
Pu236	2.64E-06	9.93E-07	3.00E-07	2.21E-07	7.41E-08	2.24E-08	2.64E-06
Pu237	3.91E-06	2.64E-18	1.78E-30	6.83E-08	4.61E-20	3.11E-32	3.91E-06
Pu238	1.86E-01	3.78E-01	3.63E-01	4.01E-02	4.84E-02	4.65E-02	3.78E-01
Pu239	6.04E+00	2.10E+01	2.10E+01	1.44E+00	1.63E+00	1.63E+00	2.10E+01
Pu240	4.26E+00	4.41E+00	4.50E+00	1.89E-01	1.89E-01	1.89E-01	4.50E+00
Pu241	2.24E+00	1.76E+00	1.38E+00	6.85E-02	5.38E-02	4.23E-02	2.24E+00
Pu242	5.25E+00	5.25E+00	5.25E+00	5.04E-03	5.04E-03	5.04E-03	5.25E+00
Pu243	9.08E-02	1.93E-15	1.93E-15	0.00E+00	0.00E+00	0.00E+00	9.08E-02
Pu244	7.91E-14	5.81E-11	1.16E-10	0.00E+00	0.00E+00	0.00E+00	1.16E-10
Pu246	1.06E-17	2.70E-18	2.70E-18	0.00E+00	0.00E+00	0.00E+00	1.06E-17
Plutonium	**1.81E+01**	**3.28E+01**	**3.25E+01**	**1.74E+00**	**1.92E+00**	**1.91E+00**	**3.28E+01**
Po210	2.36E-20	1.79E-17	2.26E-16	9.63E-21	3.13E-17	3.24E-16	3.24E-16
Po216	4.94E-21	8.34E-17	1.48E-16	1.09E-20	7.06E-18	1.20E-17	1.48E-16
Polonium	**2.86E-20**	**1.01E-16**	**3.74E-16**	**2.05E-20**	**3.83E-17**	**3.36E-16**	**3.74E-16**
Pr141	2.98E+00	7.43E+00	7.43E+00	3.55E+00	9.24E+00	9.24E+00	9.24E+00
Pr143	1.83E+00	0.00E+00	0.00E+00	3.36E+00	0.00E+00	0.00E+00	3.36E+00
Pr144	2.04E-03	3.66E-06	4.31E-08	4.53E-04	4.18E-06	4.91E-08	2.04E-03
Pr144m	1.86E-06	2.14E-08	2.51E-10	2.09E-06	2.44E-08	2.86E-10	2.09E-06
Praseodymium	**4.80E+00**	**7.43E+00**	**7.43E+00**	**6.91E+00**	**9.24E+00**	**9.24E+00**	**9.24E+00**
Pm145	1.03E-07	8.51E-08	7.00E-08	3.49E-08	2.89E-08	2.38E-08	1.03E-07
Pm146	4.16E-05	2.23E-05	1.19E-05	1.41E-05	7.56E-06	4.04E-06	4.16E-05
Pm147	3.40E-01	2.80E-01	7.46E-02	2.16E+00	8.75E-01	2.34E-01	2.16E+00
Pm148	1.12E-02	6.05E-19	0.00E+00	3.20E-02	7.09E-18	3.45E-31	3.20E-02

Radionuclide/ Element	Low Enrichment			High Enrichment			MAX
	initial	5.0 yr	10.0 yr	initial	5.0 yr	10.0 yr	
Pm148m	1.81E-03	8.80E-17	4.28E-30	2.13E-02	1.03E-15	5.01E-29	2.13E-02
Promethium	**3.53E-01**	**2.80E-01**	**7.46E-02**	**2.22E+00**	**8.75E-01**	**2.34E-01**	**2.22E+00**
Pa231	5.80E-10	1.56E-09	2.54E-09	9.18E-08	3.38E-06	6.65E-06	6.65E-06
Pa233	1.78E-08	1.69E-07	1.69E-07	8.35E-09	4.33E-08	4.33E-08	1.69E-07
Pa234	9.81E-11	1.65E-13	1.65E-13	1.88E-12	1.46E-14	1.46E-14	9.81E-11
Pa234m	6.05E-13	3.69E-13	3.69E-13	3.11E-14	3.29E-14	3.29E-14	6.05E-13
Protactinium	**1.84E-08**	**1.70E-07**	**1.71E-07**	**1.00E-07**	**3.42E-06**	**6.69E-06**	**6.69E-06**
Ra223	5.78E-18	1.63E-16	4.19E-16	4.51E-17	2.43E-13	8.94E-13	8.94E-13
Ra224	1.12E-14	1.89E-10	3.34E-10	2.46E-14	1.60E-11	2.70E-11	3.34E-10
Ra225	5.31E-16	4.64E-16	1.80E-15	3.14E-17	8.86E-16	1.99E-15	1.99E-15
Ra226	1.65E-16	1.80E-12	1.34E-11	8.78E-16	3.75E-12	1.56E-11	1.56E-11
Ra228	1.83E-19	3.88E-16	1.28E-15	2.45E-19	8.18E-16	2.71E-15	2.71E-15
Radium	**1.19E-14**	**1.91E-10**	**3.47E-10**	**2.56E-14**	**2.00E-11**	**4.35E-11**	**3.47E-10**
Rn220	1.93E-18	3.26E-14	5.76E-14	4.24E-18	2.76E-15	4.68E-15	5.76E-14
Rn222	8.59E-22	1.16E-17	8.58E-17	4.16E-21	2.41E-17	1.00E-16	1.00E-16
Radon	**1.93E-18**	**3.26E-14**	**5.77E-14**	**4.24E-18**	**2.79E-15**	**4.78E-15**	**5.77E-14**
Rh102	2.63E-06	7.94E-07	2.40E-07	7.83E-07	2.36E-07	7.16E-08	2.63E-06
Rh103	4.38E-01	3.64E+00	3.64E+00	1.10E+00	3.46E+00	3.46E+00	3.64E+00
Rh103m	3.16E-03	3.14E-17	3.10E-31	2.34E-03	2.31E-17	2.29E-31	3.16E-03
Rh106	3.93E-05	5.99E-08	1.99E-09	3.04E-06	1.45E-08	4.81E-10	3.93E-05
Rhodium	**4.41E-01**	**3.64E+00**	**3.64E+00**	**1.10E+00**	**3.46E+00**	**3.46E+00**	**3.64E+00**
Rb85	7.44E-01	8.01E-01	8.43E-01	9.34E-01	1.01E+00	1.06E+00	1.06E+00
Rb86	2.54E-03	7.94E-33	0.00E+00	1.56E-04	4.90E-34	0.00E+00	2.54E-03
Rb87	1.96E+00	1.96E+00	1.96E+00	2.51E+00	2.53E+00	2.53E+00	2.53E+00
Rubidium	**2.71E+00**	**2.76E+00**	**2.81E+00**	**3.45E+00**	**3.54E+00**	**3.59E+00**	**3.59E+00**
Ru99	1.98E-04	3.00E-04	3.85E-04	2.11E-04	3.43E-04	4.55E-04	4.55E-04
Ru100	1.64E+00	1.64E+00	1.64E+00	1.74E-01	1.74E-01	1.74E-01	1.64E+00
Ru101	5.53E+00	5.53E+00	5.53E+00	5.79E+00	5.79E+00	5.79E+00	5.79E+00
Ru102	5.79E+00	5.79E+00	5.79E+00	4.96E+00	4.96E+00	4.96E+00	5.79E+00
Ru103	3.19E+00	3.16E-14	3.13E-28	2.36E+00	2.34E-14	2.31E-28	3.19E+00
Ru104	3.78E+00	3.78E+00	3.78E+00	2.19E+00	2.19E+00	2.19E+00	3.78E+00
Ru106	1.95E+00	6.46E-02	2.14E-03	4.73E-01	1.56E-02	5.19E-04	1.95E+00
Ruthenium	**2.19E+01**	**1.68E+01**	**1.67E+01**	**1.59E+01**	**1.31E+01**	**1.31E+01**	**2.19E+01**
Sm145	6.23E-10	1.50E-11	3.64E-13	1.55E-10	3.74E-12	9.03E-14	6.23E-10
Sm146	1.44E-06	8.01E-06	1.15E-05	3.68E-07	2.60E-06	3.80E-06	1.15E-05
Sm147	7.79E-03	7.76E-01	9.81E-01	3.21E-02	2.44E+00	3.08E+00	3.08E+00
Sm148	9.49E-02	1.08E-01	1.08E-01	8.64E-02	1.40E-01	1.40E-01	1.40E-01
Sm149	7.28E-04	1.01E-01	1.01E-01	4.46E-02	1.90E-01	1.90E-01	1.90E-01
Sm150	1.86E+00	1.88E+00	1.88E+00	1.95E+00	1.95E+00	1.95E+00	1.95E+00
Sm151	2.16E-02	4.61E-02	4.44E-02	1.31E-01	1.48E-01	1.43E-01	1.48E-01
Sm152	7.83E-01	7.83E-01	7.83E-01	9.16E-01	9.16E-01	9.16E-01	9.16E-01
Sm154	1.04E+00	1.04E+00	1.04E+00	1.36E-01	1.36E-01	1.36E-01	1.04E+00
Samarium	**3.81E+00**	**4.73E+00**	**4.93E+00**	**3.30E+00**	**5.92E+00**	**6.55E+00**	**6.55E+00**
Se76	2.39E-05	2.78E-05	2.78E-05	2.45E-06	2.63E-06	2.63E-06	2.78E-05
Se77	4.73E-03	4.95E-03	4.95E-03	6.98E-03	7.34E-03	7.34E-03	7.34E-03
Se78	2.05E-02	2.06E-02	2.06E-02	1.94E-02	1.95E-02	1.95E-02	2.06E-02
Se79	3.56E-02	3.56E-02	3.56E-02	3.98E-02	3.99E-02	3.99E-02	3.99E-02
Se80	1.05E-01	1.05E-01	1.05E-01	1.20E-01	1.20E-01	1.20E-01	1.20E-01
Se82	2.56E-01	2.56E-01	2.56E-01	3.05E-01	3.05E-01	3.05E-01	3.05E-01

Radionuclide/ Element	Low Enrichment			High Enrichment			MAX
	initial	5.0 yr	10.0 yr	initial	5.0 yr	10.0 yr	
Selenium	**4.22E-01**	**4.22E-01**	**4.22E-01**	**4.91E-01**	**4.91E-01**	**4.91E-01**	**4.91E-01**
Si28	2.63E+00	2.63E+00	2.63E+00	1.48E-01	1.48E-01	1.48E-01	2.63E+00
Si29	2.36E-03	2.36E-03	2.36E-03	6.90E-06	6.90E-06	6.90E-06	2.36E-03
Si30	2.31E-06	2.31E-06	2.31E-06	3.36E-10	3.36E-10	3.36E-10	2.31E-06
Si32	2.66E-14	2.60E-14	2.55E-14	0.00E+00	0.00E+00	0.00E+00	2.66E-14
Silicon	**2.63E+00**	**2.63E+00**	**2.63E+00**	**1.48E-01**	**1.48E-01**	**1.48E-01**	**2.63E+00**
Ag107	4.91E-09	7.33E-07	1.46E-06	1.29E-09	9.59E-08	1.90E-07	1.46E-06
Ag108	2.75E-10	3.79E-15	3.69E-15	2.39E-12	6.70E-17	6.53E-17	2.75E-10
Ag108m	1.26E-06	1.23E-06	1.19E-06	2.23E-08	2.18E-08	2.11E-08	1.26E-06
Ag109	4.14E-01	4.60E-01	4.60E-01	4.30E-02	4.39E-02	4.39E-02	4.60E-01
Ag109m	3.70E-05	6.89E-17	4.46E-18	7.14E-07	4.63E-19	3.00E-20	3.70E-05
Ag110	1.66E-05	1.70E-12	1.07E-14	9.18E-08	1.90E-14	1.19E-16	1.66E-05
Ag110m	1.74E-02	1.10E-04	6.90E-07	1.95E-04	1.22E-06	7.70E-09	1.74E-02
Ag111	6.54E-02	0.00E+00	0.00E+00	5.53E-03	0.00E+00	0.00E+00	6.54E-02
Silver	**4.97E-01**	**4.60E-01**	**4.60E-01**	**4.87E-02**	**4.39E-02**	**4.39E-02**	**4.97E-01**
Na23	5.36E-10	5.36E-10	5.36E-10	1.98E-11	1.98E-11	1.98E-11	5.36E-10
Sodium	**5.36E-10**	**5.36E-10**	**5.36E-10**	**1.98E-11**	**1.98E-11**	**1.98E-11**	**5.36E-10**
Sr86	9.95E-04	3.54E-03	3.54E-03	1.04E-04	2.61E-04	2.61E-04	3.54E-03
Sr87	1.48E-05	1.50E-05	1.50E-05	4.41E-06	4.41E-06	4.41E-06	1.50E-05
Sr88	2.86E+00	2.86E+00	2.86E+00	3.64E+00	3.65E+00	3.65E+00	3.65E+00
Sr89	2.60E+00	3.46E-11	4.63E-22	3.56E+00	4.75E-11	6.34E-22	3.56E+00
Sr90	4.64E+00	4.10E+00	3.63E+00	6.03E+00	5.33E+00	4.70E+00	6.03E+00
Strontium	**1.01E+01**	**6.97E+00**	**6.49E+00**	**1.32E+01**	**8.98E+00**	**8.35E+00**	**1.32E+01**
S32	5.16E-14	3.75E-13	3.75E-13	4.15E-20	1.54E-19	1.54E-19	3.75E-13
S33	4.03E-17	4.11E-17	4.11E-17	0.00E+00	0.00E+00	0.00E+00	4.11E-17
Sulfur	**5.17E-14**	**3.75E-13**	**3.75E-13**	**4.15E-20**	**1.54E-19**	**1.54E-19**	**3.75E-13**
Tc98	6.04E-05	6.04E-05	6.04E-05	1.01E-05	1.01E-05	1.01E-05	6.04E-05
Tc99	4.65E+00	5.19E+00	5.19E+00	6.28E+00	6.84E+00	6.84E+00	6.84E+00
Technetium	**4.65E+00**	**5.19E+00**	**5.19E+00**	**6.28E+00**	**6.84E+00**	**6.84E+00**	**6.84E+00**
Te122	1.10E-03	1.50E-03	1.50E-03	7.38E-05	8.74E-05	8.74E-05	1.50E-03
Te123	4.23E-06	9.73E-06	9.73E-06	6.20E-08	9.33E-08	9.33E-08	9.73E-06
Te123m	5.50E-06	1.40E-10	3.56E-15	3.13E-08	7.96E-13	2.03E-17	5.50E-06
Te124	1.25E-04	1.19E-03	1.19E-03	1.64E-05	8.81E-05	8.81E-05	1.19E-03
Te125	6.26E-04	4.68E-02	5.99E-02	5.10E-04	3.03E-02	3.88E-02	5.99E-02
Te125m	1.56E-04	2.56E-04	7.20E-05	1.21E-04	1.65E-04	4.64E-05	2.56E-04
Te126	1.39E-03	1.91E-03	1.93E-03	3.38E-04	4.05E-04	4.08E-04	1.93E-03
Te127	4.50E-03	1.44E-09	1.30E-14	1.75E-03	8.86E-10	8.01E-15	4.50E-03
Te127m	3.60E-02	4.11E-07	3.71E-12	2.44E-02	2.54E-07	2.29E-12	3.60E-02
Te128	6.15E-01	6.16E-01	6.16E-01	5.14E-01	5.14E-01	5.14E-01	6.16E-01
Te129	1.91E-03	6.31E-21	0.00E+00	1.30E-03	5.05E-21	0.00E+00	1.91E-03
Te129m	1.56E-01	6.84E-18	2.96E-34	1.25E-01	5.48E-18	2.38E-34	1.56E-01
Te130	2.66E+00	2.66E+00	2.66E+00	2.60E+00	2.60E+00	2.60E+00	2.66E+00
Tellurium	**3.48E+00**	**3.33E+00**	**3.34E+00**	**3.27E+00**	**3.14E+00**	**3.15E+00**	**3.48E+00**
Tb159	2.78E-02	3.15E-02	3.15E-02	1.85E-03	1.90E-03	1.90E-03	3.15E-02
Tb160	1.25E-03	3.11E-11	7.74E-19	2.25E-05	5.59E-13	1.39E-20	1.25E-03
Tb161	2.91E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.91E-03
Terbium	**3.19E-02**	**3.15E-02**	**3.15E-02**	**1.87E-03**	**1.90E-03**	**1.90E-03**	**3.19E-02**
Tl207	0.00E+00	0.00E+00	0.00E+00	1.21E-20	6.49E-17	2.40E-16	2.40E-16
Tl208	2.16E-18	3.65E-14	6.46E-14	4.75E-18	3.09E-15	5.23E-15	6.46E-14

Radionuclide/ Element	Low Enrichment			High Enrichment			MAX
	Initial	5.0 yr	10.0 yr	Initial	5.0 yr	10.0 yr	
Thallium	**2.16E-18**	**3.65E-14**	**6.46E-14**	**4.76E-18**	**3.15E-15**	**5.47E-15**	**6.46E-14**
Th227	1.68E-17	2.66E-16	6.88E-16	1.48E-16	3.98E-13	1.46E-12	1.46E-12
Th228	2.86E-12	3.66E-08	6.49E-08	6.55E-12	3.10E-09	5.25E-09	6.49E-08
Th229	1.90E-12	9.16E-11	3.55E-10	2.35E-12	1.75E-10	3.93E-10	3.93E-10
Th230	2.19E-10	1.14E-07	4.31E-07	2.06E-09	1.69E-07	3.61E-07	4.31E-07
Th231	1.84E-11	8.15E-13	8.26E-13	2.80E-09	2.75E-09	2.75E-09	2.80E-09
Th232	7.00E-08	3.81E-06	7.55E-06	1.10E-07	8.08E-06	1.60E-05	1.60E-05
Th234	9.11E-09	1.09E-08	1.09E-08	7.54E-10	9.74E-10	9.74E-10	1.09E-08
Thorium	**7.94E-08**	**3.97E-06**	**8.06E-06**	**1.16E-07**	**8.25E-06**	**1.64E-05**	**1.64E-05**
Tm169	8.41E-09	1.71E-08	1.71E-08	2.88E-10	4.21E-10	4.21E-10	1.71E-08
Tm170	9.65E-11	5.11E-15	2.71E-19	8.59E-13	4.55E-17	2.41E-21	9.65E-11
Tm171	7.89E-09	1.40E-09	2.30E-10	9.28E-10	1.54E-10	2.54E-11	7.89E-09
Thulium	**1.64E-08**	**1.85E-08**	**1.74E-08**	**1.22E-09**	**5.75E-10**	**4.47E-10**	**1.85E-08**
Sn114	5.03E-08	1.45E-07	1.45E-07	2.31E-09	5.91E-09	5.91E-09	1.45E-07
Sn115	8.29E-04	9.75E-04	9.75E-04	6.60E-04	7.08E-04	7.08E-04	9.75E-04
Sn116	1.59E-02	1.60E-02	1.60E-02	1.40E-03	1.40E-03	1.40E-03	1.60E-02
Sn117	2.73E-02	2.76E-02	2.76E-02	1.44E-02	1.45E-02	1.45E-02	2.76E-02
Sn117m	2.89E-05	0.00E+00	0.00E+00	1.01E-05	0.00E+00	0.00E+00	2.89E-05
Sn118	2.33E-02	2.33E-02	2.33E-02	1.48E-02	1.48E-02	1.48E-02	2.33E-02
Sn119	2.43E-02	2.44E-02	2.44E-02	1.65E-02	1.65E-02	1.65E-02	2.44E-02
Sn119m	9.06E-05	1.21E-06	1.60E-08	5.78E-05	7.69E-07	1.02E-08	9.06E-05
Sn120	2.46E-02	2.46E-02	2.46E-02	1.66E-02	1.66E-02	1.66E-02	2.46E-02
Sn121	1.28E-03	9.83E-09	9.23E-09	5.49E-04	4.81E-09	4.51E-09	1.28E-03
Sn121m	2.40E-04	2.26E-04	2.11E-04	1.18E-04	1.11E-04	1.04E-04	2.40E-04
Sn122	3.14E-02	3.14E-02	3.14E-02	2.14E-02	2.14E-02	2.14E-02	3.14E-02
Sn123	2.31E-03	1.28E-07	7.09E-12	2.14E-03	1.18E-07	6.56E-12	2.31E-03
Sn124	5.33E-02	5.33E-02	5.33E-02	3.68E-02	3.68E-02	3.68E-02	5.33E-02
Sn125	6.34E-03	0.00E+00	0.00E+00	3.56E-03	0.00E+00	0.00E+00	6.34E-03
Sn126	1.19E-01	1.19E-01	1.19E-01	6.30E-02	6.30E-02	6.30E-02	1.19E-01
Tin	**3.30E-01**	**3.20E-01**	**3.20E-01**	**1.92E-01**	**1.86E-01**	**1.86E-01**	**3.30E-01**
Ti47	4.49E-19	7.11E-19	7.11E-19	0.00E+00	0.00E+00	0.00E+00	7.11E-19
Ti48	1.20E-16	1.58E-16	1.58E-16	5.56E-18	6.28E-18	6.28E-18	1.58E-16
Ti49	5.89E-13	5.89E-13	5.89E-13	2.30E-14	2.30E-14	2.30E-14	5.89E-13
Ti50	8.75E-13	8.75E-13	8.75E-13	2.63E-15	2.63E-15	2.63E-15	8.75E-13
Titanium	**1.46E-12**	**1.46E-12**	**1.46E-12**	**2.56E-14**	**2.56E-14**	**2.56E-14**	**1.46E-12**
U232	1.26E-08	2.20E-06	2.75E-06	1.65E-08	1.80E-07	2.20E-07	2.75E-06
U233	1.89E-08	8.29E-06	1.63E-05	7.01E-06	9.14E-06	1.12E-05	1.63E-05
U234	6.98E-04	1.56E-02	3.00E-02	1.10E-02	1.30E-02	1.48E-02	3.00E-02
U235	1.98E-01	2.00E-01	2.04E-01	6.78E+02	6.78E+02	6.78E+02	6.78E+02
U236	2.58E+01	2.58E+01	2.58E+01	5.48E+01	5.48E+01	5.48E+01	5.48E+01
U237	3.15E+00	5.33E-08	4.19E-08	4.40E-01	1.63E-09	1.28E-09	3.15E+00
U238	7.53E+02	7.53E+02	7.53E+02	6.70E+01	6.70E+01	6.70E+01	7.53E+02
Uranium	**7.82E+02**	**7.78E+02**	**7.78E+02**	**8.00E+02**	**7.99E+02**	**7.99E+02**	**8.00E+02**
V50	8.83E-16	8.83E-16	8.83E-16	4.31E-17	4.31E-17	4.31E-17	8.83E-16
V51	4.71E-09	4.71E-09	4.71E-09	9.68E-10	9.68E-10	9.68E-10	4.71E-09
Vanadium	**4.71E-09**	**4.71E-09**	**4.71E-09**	**9.68E-10**	**9.68E-10**	**9.68E-10**	**4.71E-09**
Yb170	6.04E-12	1.02E-10	1.02E-10	7.33E-14	9.31E-13	9.31E-13	1.02E-10
Yb171	1.79E-10	7.29E-09	8.46E-09	2.14E-11	8.04E-10	9.34E-10	8.46E-09
Yb172	1.23E-08	1.84E-08	1.84E-08	7.53E-10	8.84E-10	8.84E-10	1.84E-08

Radionuclide/ Element	Low Enrichment			High Enrichment			MAX
	initial	5.0 yr	10.0 yr	initial	5.0 yr	10.0 yr	
Ytterbium	**1.24E-08**	**2.58E-08**	**2.69E-08**	**7.74E-10**	**1.69E-09**	**1.82E-09**	**2.69E-08**
Y89	1.20E+00	3.80E+00	3.80E+00	1.38E+00	4.94E+00	4.94E+00	4.94E+00
Y89m	1.68E-08	1.19E-20	1.58E-31	2.78E-09	1.63E-20	2.16E-31	1.68E-08
Y90	4.09E-03	1.07E-03	9.43E-04	1.54E-03	1.39E-03	1.22E-03	4.09E-03
Y91	3.53E+00	1.43E-09	5.70E-19	4.58E+00	1.86E-09	7.45E-19	4.58E+00
Yttrium	**4.73E+00**	**3.80E+00**	**3.80E+00**	**5.95E+00**	**4.94E+00**	**4.94E+00**	**5.95E+00**
Xe126	2.05E-11	2.05E-11	2.05E-11	8.89E-12	8.89E-12	8.89E-12	2.05E-11
Xe127	1.46E-08	1.15E-23	0.00E+00	1.19E-10	9.35E-26	0.00E+00	1.46E-08
Xe128	1.02E-02	1.03E-02	1.03E-02	5.55E-04	5.56E-04	5.56E-04	1.03E-02
Xe129	9.11E-05	1.41E-04	1.43E-04	3.03E-07	6.48E-07	8.89E-07	1.43E-04
Xe129m	5.05E-05	0.00E+00	0.00E+00	1.04E-07	0.00E+00	0.00E+00	5.05E-05
Xe130	2.00E-01	2.10E-01	2.10E-01	1.00E-02	1.03E-02	1.03E-02	2.10E-01
Xe131	1.29E+00	2.44E+00	2.44E+00	3.04E+00	4.05E+00	4.05E+00	4.05E+00
Xe131m	2.11E-02	0.00E+00	0.00E+00	1.43E-02	0.00E+00	0.00E+00	2.11E-02
Xe132	8.14E+00	8.78E+00	8.78E+00	6.16E+00	6.78E+00	6.78E+00	8.78E+00
Xe134	1.36E+01	1.36E+01	1.36E+01	1.21E+01	1.21E+01	1.21E+01	1.36E+01
Xe136	1.98E+01	1.98E+01	1.98E+01	1.93E+01	1.93E+01	1.93E+01	1.98E+01
Xenon	**4.30E+01**	**4.48E+01**	**4.48E+01**	**4.05E+01**	**4.22E+01**	**4.22E+01**	**4.48E+01**
Zn66	3.91E-09	4.40E-09	4.40E-09	2.23E-09	2.38E-09	2.38E-09	4.40E-09
Zn67	5.30E-10	5.90E-10	5.90E-10	3.99E-10	4.30E-10	4.30E-10	5.90E-10
Zn68	4.81E-11	4.81E-11	4.81E-11	2.00E-11	2.00E-11	2.00E-11	4.81E-11
Zn70	4.61E-08	4.61E-08	4.61E-08	1.38E-08	1.38E-08	1.38E-08	4.61E-08
Zinc	**5.06E-08**	**5.12E-08**	**5.12E-08**	**1.64E-08**	**1.66E-08**	**1.66E-08**	**5.12E-08**
Zr90	1.58E-02	5.56E-01	1.03E+00	9.15E-03	7.06E-01	1.33E+00	1.33E+00
Zr91	1.34E+00	4.89E+00	4.89E+00	1.50E+00	6.14E+00	6.14E+00	6.14E+00
Zr92	5.15E+00	5.16E+00	5.16E+00	6.20E+00	6.25E+00	6.25E+00	6.25E+00
Zr93	3.65E+00	3.68E+00	3.68E+00	4.40E+00	4.45E+00	4.45E+00	4.45E+00
Zr94	6.10E+00	6.10E+00	6.10E+00	6.90E+00	6.90E+00	6.90E+00	6.90E+00
Zr95	4.68E+00	1.21E-08	3.11E-17	5.40E+00	1.40E-08	3.60E-17	5.40E+00
Zr96	6.24E+00	6.24E+00	6.24E+00	6.84E+00	6.84E+00	6.84E+00	6.84E+00
Zirconium	**2.72E+01**	**2.66E+01**	**2.71E+01**	**3.12E+01**	**3.13E+01**	**3.19E+01**	**3.19E+01**

As a result of the concern for volatilization of the radionuclide species listed in Table A.9 during melt-dilute treatment, the available vapor pressure data for the radionuclide species present in quantities greater than 10^{-3} grams has been compiled in Table A.10. Table A.10 provides the temperature dependence of the vapor pressure for only the elemental radionuclide species—no compound formation between radionuclides or other fuel species is included. However, Table A.11 displays the possible compounds formed between Al, U, Cs, Rb, Te, Ba, Sr, I, and Tritium.

Table A.10 Vapor Pressure in Atmospheres

Element	Temperature, °C					
	600	700	800	900	Range	T (mp)
Aluminum	2.20E-13	1.78E-11	6.35E-10	1.23E-08	660 to1800	660
Americium (sol)	1.28E-10	6.54E-09	1.59E-07	2.21E-06	298 to mp	994
Barium (liq)	4.53E-06	4.14E-05	2.51E-04	1.12E-03	mp to 1200	725
Cadmium	1.16E-01	5.02E-01	1.65E+00	4.42E+00	mp to 650	321
Cerium	2.12E-19	6.65E-17	7.13E-15	3.45E-13	mp to 2450	798
Cesium	6.00E-01	1.69E+00	3.94E+00	7.94E+00	mp to 550	28
Curium	1.86E-16	2.65E-14	1.50E-12	4.25E-11	mp to 2200	1340
Dysprosium (sol)	5.54E-12	3.14E-10	8.28E-09	1.24E-07	298 to mp	1412
Europium (sol)	9.45E-06	1.08E-04	7.77E-04	3.95E-03	298 to mp	822
Gadolinium (sol)	2.23E-17	4.27E-15	3.07E-13	1.07E-11	298 to mp	1313
Iron	8.42E-17	1.70E-14	1.27E-12	4.57E-11	mp to 2100	1535
Lanthanum	7.53E-20	2.82E-17	3.49E-15	1.90E-13	mp to 2450	918
Magnesium (sol)	1.29E-03	9.84E-03	5.09E-02	1.97E-01	298 to mp	649
Manganese (sol)	1.77E-10	8.74E-09	2.05E-07	2.77E-06	298 to mp	1244
Molybdenum	3.41E-32	3.60E-28	6.68E-25	3.40E-22	298 to 2500	2617
Neodymium	6.11E-14	4.46E-12	1.46E-10	2.64E-09	mp to 2000	1021
Neptunium	2.51E-21	1.23E-18	1.87E-16	1.20E-14	mp to 2500	640
Niobium (sol)	5.56E-36	1.53E-31	6.26E-28	6.18E-25	298 to 2500	2468
Palladium	8.38E-16	1.07E-13	5.56E-12	1.47E-10	mp to 2100	1554
Plutonium	3.84E-16	3.51E-14	1.38E-12	2.92E-11	mp to 2450	641
Rhodium	1.30E-24	1.85E-21	6.80E-19	9.15E-17	mp to 2500	1965
Samarium						1074
Selenium	3.11E-01	1.20E+00	3.62E+00	9.02E+00	mp to bp	217
Silicon	6.85E-14	6.32E-12	2.48E-10	5.18E-09	mp to 3000	1410
Silver (sol)	4.35E-11	2.33E-09	5.90E-08	8.56E-07	298 to mp	962
Silver	8.19E-11	3.48E-09	7.34E-08	9.21E-07	mp to 1600	962
Strontium (sol)	8.34E-26	8.34E-26	8.34E-26	8.34E-26	298 to mp	769
Uranium (sol)	3.35E-41	4.59E-38	1.60E-35	2.00E-33	298 to mp	1132
Uranium (liq)	5.32E-25	8.33E-22	3.18E-19	4.27E-17	mp to 2500	1132
Yttrium (sol)	4.21E-19	1.62E-16	2.03E-14	1.11E-12	298 to mp	1552
Yttrium (liq)	3.13E-18	7.75E-16	6.88E-14	2.84E-12	mp to 2300	1552
Zirconium (sol)	3.90E-29	1.84E-25	1.77E-22	5.27E-20	298 to mp	1852
Zirconium (liq)	1.27E-28	4.68E-25	3.73E-22	9.53E-20	mp tp 2500	1852

Table A.11 Compounds and Alloys of Various Offgas Species with Uranium and Aluminum

	Al	U	Cs	Rb	Te	H^3	Ba	Sr	I
Al		Various solubility of both elements (see phase diagram); Al_4U (ε); Al_3U (ζ); Al_2U (η)	Slight solubility of Cs in Al; no other phases	No phase diagram available but no compounds have been observed	Limited solubility of both elements; Al_5Te; Al_2Te_3	Limited (~ zero) solubility of H in Al; no other phases	Limited solubility of Ba in Al but 5.4 at% solubility of Al in Ba; Al_4Ba; Al_2Ba; AlBa; $AlBa_2$ phases	Limited solubility of both elements; Al_4Sr; Al_2Sr, Al_2Sr_3	No phase diagram available
U	See Al-U		No phase diagram available	No phase diagram available	See Te-U	See H^3-U	No phase diagram available	No phase diagram available	No phase diagram available
Cs	See Al-Cs	No phase diagram available		Complete immiscibility of both elements	Limited solubility of both elements; Cs_2Te; Cs_2Te_2; Cs_5Te_4; CsTe; Cs_2Te_3; Cs_2Te_5; $CsTe_5$	No phase diagram available; CsH is formed	See Ba-Cs	Complete immiscibility of both elements but no phase diagram given	Limited solubility of both elements; CsI; CsI_3; CsI_4
Rb	See Al-Rb	No phase diagram available	See Cs-Rb		Limited solubility of both elements; Rb_2Te; Rb_3Te_2; Rb_5Te_4; RbTe; Rb_2Te_3; Rb_2Te_5	See H^3-Rb	See Ba-Rb	Complete immiscibility of both elements but no phase diagram given	See I-Rb
Te	See Al-Te	Limited solubility of both elements; UTe; U_3Te_4; U_3Te_5; UTe_2; U_3Te_7; U_2Te_3; UTe_3; UTe_5	See Cs-Te	See Rb-Te		No phase diagram available	See Ba-Te	See Sr-Te	See I-Te
H^3	See Al-H^3	Limited (~0 to 28 wppm) solubility of H in various U allotropes; α, β, δ, ε-UH_3	See Cs-H^3	No phase diagram available; RbH is formed	No phase diagram available		See Ba-H^3	43.2 at% solubility of H in β-Sr; 4.3 at.% solubility of H in α-Sr; α and β-SrH_2	No phase diagram available; HI may be possible but no reference has been found
Ba	See Al-Ba	No phase diagram available	Complete immiscibility of both elements but no phase diagram given	Complete immiscibility of both elements but no phase diagram given	Limited solubility of both elements; BaTe; Ba_2Te_3; $BaTe_2$	Large solubility of H in Ba (57 at.% at 950°C); α-BaH_2; β-BaH_2		Two solid solutions formed; no other phases	Limited solubility of both elements; BaI_2
Sr	See Al-Sr	No phase diagram available but no compounds have been observed	See Cs-Sr	See Rb-Sr	Limited solubility of both elements; SrTe; Sr_2Te_3; $SrTe_2$	See H^3-Sr	See Ba-Sr		See I-Sr
I	No phase diagram available	No phase diagram available	See Cs-I	Limited solubility of both elements; RbI; RbI_3	Limited solubility of both elements; TeI; TeI_4	See H^3-I	See Ba-I	Limited solubility of both elements; SrI_2	

A.5 References

1 Duguid, J. O., J. A. McNeish, V. Vallikat, D. Cresap, and N. J. Erb. *Total System Performance Assessment Sensitivity Studies of U.S. Department of Energy Spent Nuclear Fuel*, DI A00000000-01717-5705-00017, Rev. 01 (September 30, 1997).

2 Final Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Environmental Restoration and Waste Management Programs Environmental Impact Statement. DOE (U.S. Department of Energy), DOE/EIS-0203, Idaho Operations Office, Idaho Falls, Idaho (1995).

3 H. M. Brooks and R. L. Sindelar, "Characterization of FRR SNF in Basin and Dry Storage Systems," to be presented at the Third Topical Meeting on DOE Spent Nuclear Fuel and Fissile Material Management, September 1998, Charleston, SC and published in the proceeding by the American Nuclear Society.

4 S. D. Burke to R. J. Skwarek, "Proposed New Criteria for Acceptance of Al-SNF for SRS Basin Storage (U)," SRS-RSE-970167, February 11. 1998.

WESTINGHOUSE SAVANNAH RIVER CO. REPORT WSRC-TR-98-00228

DISTRIBUTION

SAVANNAH RIVER SITE

W. Poulson, 704-C
S. Wood, 773-A
G. T. Wright, 773-A
G. H. Clare, 704-C
M. W. Barlow, 704-C
E. R. Conatser, 704-C
M. E. Dupont, 707-C
R. J. Skwarek, 704-C
H. Brooks, 707-C
W. F. Swift, 707-C
S. D. Burke, 707-C
D. Bickley, 707-C
S. Large, 707-C
G. Reynolds, 704-C
T. J. Worrell, 705-K
C. R. Wolfe, 773-A
J. D. Cohen, 773-A
T. L. Capeletti, 773-41A
C. Holding-Smith, 773-42A
N. C. Iyer, 773-A
M. R. Louthan, Jr., 773-A
J. F. Krupa, 773-41A
J. P. Howell, 773-41A
H. B. Peacock, Jr., 773-A
D. R. Leader, 773-A
F. C. Rhode, 773-A
D. L. Fisher, 773-41A
T. M. Adams, 773-41A
M. E. Hodges, 773-41A
P. J. French, 773-23A
S. Y. Lee, 773-42A
P. E. Zapp, 773-A
B. Cederdahl, 707-C
L. Lamm, 707-C
W. Bugher, 707-C
A. Brewer, 707-C
J. I. Mickalonis, 773-A
P. S. Lam, 773-41A
W. Epling, 707-C
B. J. Wiersma, 773-A
G. T. Chandler, 773-A
W. F Ayres, 773-41A
R. L. Sindelar, 773-41A
D. W. Vinson, 773-41A
DOE-SR (10 copies)/J. M. Ridley, 707-K
Site Records, 773-52A